March 25, 2002

Mr. W. E. Cummins, Director
Advanced Plant Development Unit
Westinghouse Electric Company
Post Office Box 355
Pittsburgh, Pennsylvania 15230-0355

SUBJECT: APPLICABILITY OF AP600 STANDARD PLANT DESIGN ANALYSIS CODES, TEST PROGRAM AND EXEMPTIONS TO THE AP1000 STANDARD PLANT DESIGN

Westinghouse and the Nuclear Regulatory Commission (NRC) agreed to a 3-phase approach for the AP1000 design certification application. The objectives of the Phase 1 review were to identify the review assumptions and issues to be reviewed in the Phase 2 pre-application review. The objective of the Phase 2 review was to resolve those identified issues or identify issues to be resolved in Phase 3. Phase 3 will be the review of the design certification application for the AP1000 where the actual design and safety analyses will be submitted and evaluated.

By letter dated May 4, 2000, Westinghouse requested that the NRC proceed with the Phase 1 assessment to identify the issues that the staff would evaluate during Phase 2. The staff documented its completion of the Phase 1 review in its letter dated July 27, 2000.

By letter dated August 28, 2000, Westinghouse requested that the NRC proceed with Phase 2 of the AP1000 pre-application review to address the following four issues:

(1) applicability of the AP600 test program to the AP1000 design,
(2) applicability of the AP600 analysis codes to the AP1000 design,
(3) use of additional design acceptance criteria (DAC), and
(4) use of certain exemptions for the AP1000 design.

This correspondence sets forth, as detailed in the enclosure, the staff's conclusions and recommendations regarding the applicability of the AP600 test program and analysis codes to the AP1000 design. This letter also documents the staff's assessment of Westinghouse's proposal to request certain exemptions for the AP1000 design certification. The staff's position is founded on evaluations of the proposed AP1000 design, a significant exchange of information with Westinghouse, and the staff's experience with the certification review of the Westinghouse AP600 standard plant design.

The staff's assessment regarding Westinghouse's proposed use of DAC in the instrumentation and controls, control room (human factors engineering) and piping design areas will be documented in a paper to the Commission.

Applicability of AP600 Test Program and Analysis Codes to the AP1000 Standard Design

The enclosure to this letter presents the staff's review of the Westinghouse reports regarding the applicability of the AP600 analysis codes and test programs to the AP1000 design. In summary, the staff finds that, with some exceptions, the experimental data produced by the AP600 separate-effects and integral-system test programs are appropriate for use in support of the AP1000 analysis, and the analysis codes validated for the AP600 design could be extended to the analysis of the AP1000 design. The most significant technical issue of concern is the behavior of Stage 4 of the automatic depressurization system (ADS-4). Resolution of this issue will require the successful validation of certain analytical models employed in the proposed computer codes. The exceptions are as follows:

- Westinghouse has not demonstrated that the existing AP600 integral tests provide data over the range of conditions necessary to validate entrainment models in the NOTRUMP and WCOBRA/TRAC codes that they intend to use. In particular, the NOTRUMP code lacks acceptable models for liquid entrainment in the upper plenum or from a horizontal stratified water level in the hot legs during the ADS-4 actuation.

- The review of the ability of the LOFTRAN code to evaluate potential steam voids within the reactor system following a main steamline break (MSLB) will be deferred to Phase 3, since Westinghouse did not provide an MSLB analysis for the AP1000 plant design.

- Westinghouse needs to qualify the penalty factor used with the NOTRUMP passive residual heat removal (PRHR) heat exchanger model. Existing PRHR heat exchanger test data show the boiling heat transfer correlation used in NOTRUMP to be non-conservative at high heat fluxes. The difference between the correlation predictions and test data becomes significant for the PRHRHX heat fluxes predicted for the AP1000, which are larger than those predicted for the AP600 standard plant design.

- Westinghouse did not justify that the increased flow area of the ADS-4 would support the liquid expulsion to avoid boron precipitation in the vessel during long-term cooling.

- Westinghouse did not justify the methodology used to calculate peak clad temperature (PCT) in the event that the core becomes uncovered during a small break LOCA.

- Westinghouse did not properly scale the containment large scale test (LST) for transients, and the test is only valid for steady-state conditions. This limitation was identified during the AP600 review and also applies to the AP1000 design. However, the LST does support the mass and heat transfer correlations used in the WGOTHIC code for the AP600 and the AP1000. Westinghouse needs to perform the WGOTHIC containment analyses with an evaluation model and appropriate boundary conditions to ensure that the mass and heat transfer correlations remain valid for the AP1000 design.

Severe accident thermal-hydraulics were not part of the Phase 2 review, and will be addressed in Phase 3. Therefore, the staff review and assessment of scaling did not address containment phenomena and was limited to those affecting the AP1000 primary system. The review and assessment of the applicability of the AP600 test programs to the AP1000 design considered both the primary system and the containment.

Applicability of the Proposed Exemption Requests for the AP1000 Standard Design

For the AP1000 design, Westinghouse has proposed to request exemptions from the regulations for the plant safety parameter display console (10 CFR 50.34(f)(2)(iv)), the auxiliary (or emergency) feedwater system (10 CFR 50.62(c)(1)), and offsite power sources (10 CFR Part 50 Appendix A, General Design Criterion (GDC) 17).

Section 50.34(f)(2)(iv) of the NRC's regulations requires a "safety parameter display console that will display to operators a minimum set of parameters defining the safety status..., displaying a full range of important plant parameters..., and capable of indicating when process limits are being approached or exceeded." The AP600 design certification includes a DAC which is based on the conclusion that the safety parameter display functions of advanced control rooms should be integrated into the main control room design. Thus, the exemption criterion of 10 CFR 50.12(a)(2)(ii), that an exemption may be granted if the "application of the regulation ... is not necessary to achieve the underlying purpose of the rule...," was met. Westinghouse proposes to replicate the AP600 control room design for the AP1000. The staff agrees that, if no new issues arise, the underlying purpose is still met, and that the request for an exemption similar to that granted for the AP600 design should be acceptable for the AP1000 design.

Section 50.62(c)(1) of the NRC's regulations requires that equipment be available to ensure the automatic startup of the auxiliary feedwater (AFW) system under anticipated transient without scram (ATWS) conditions. For current and evolutionary plant designs, the regulation requires an AFW system. The AP600 design met the requirement for emergency core cooling through its passive residual heat removal (PRHR) system, which is initiated automatically under the conditions of an ATWS. The AP1000 design also has a PRHR system similar to that of the AP600 design, which is re-engineered for the larger plant. The staff agrees that this design feature appears to satisfy the underlying purpose of the regulation, and thus could meet the exemption criterion of 10 CFR 50.12(a)(2)(ii).

GDC 17 of 10 CFR Part 50, Appendix A, requires two physically independent offsite power sources. The AP600 design was based on safety-related passive systems for core cooling and containment integrity, which did not rely on offsite power sources. In this regard, the AP1000 design should also preclude the need to rely on offsite power sources, because it is also based on safety-related passive systems for core cooling and containment integrity. The staff agrees that this similarity of the AP1000 design to the AP600 design appears to satisfy the underlying purpose of the regulation, and thus could meet the exemption criterion of 10 CFR 50.12(a)(2)(ii).

The staff has reviewed Westinghouse's proposals for exemptions to the Commission's regulations, and concludes that, given the current understanding of the similarity between the AP600 and AP1000 designs, the proposed exemptions are applicable and are expected to be justifiable. This conclusion is contingent on the extent to which potential dissimilarities between the AP600 and AP1000 designs affect the safety areas involved.

Conclusions

The staff concludes that Westinghouse's test program for the AP600 design is sufficient for validating analysis codes necessary for development of the AP1000 design subject to resolution of the exceptions noted. Further, the computer codes used in the analysis of accidents, transients, and containment performance for the AP600 design are applicable to the AP1000 design subject to resolution of the exceptions noted in the assessment. In particular, liquid entrainment modeling is important in demonstrating the success of passive emergency core cooling system (ECCS) operation; however, there is currently insufficient experimental data to validate the entrainment modeling during ADS-4 actuation for the AP1000. Resolution of this issue and the others noted will be incorporated into the staff review of the AP1000 design certification (Phase 3).

The staff concludes that, given the current understanding of the similarity between the AP600 and AP1000 designs, the exemptions Westinghouse intends to request, as identified during the staff's pre-application review of the AP1000, are applicable and are expected to be justifiable.

Sincerely,

/RA/

James E. Lyons, Director
New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation

Project No. 711

Attachment: U. S. Nuclear Regulatory Commission Staff Assessment of the Applicability of the AP600 Passive Safety Systems Analysis Codes and Test Program to the AP1000 Standard Plant Design

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cc:

Mr. W. Edward Cummins
Advanced Plant Safety & Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

Mr. Michael Corletti
Advanced Plant Safety & Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

Mr. H. A. Sepp
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230

Lynn Connor
Doc-Search Associates
2211 sw 1ST Ave - #1502
Portland, OR 97201

Barton Z. Cowan, Esq.
Eckert Seamans Cherin & Mellott, LLC
600 Grant Street 44th Floor
Pittsburgh, PA 15219

Mr. Ed Rodwell, Manager
PWR Design Certification
Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94304-1395

Charles Brinkman, Director
Washington Operations
Westinghouse Electric Company
12300 Twinbrook Parkway, Suite 330
Rockville, MD 20852

Mr. R. Simard
Nuclear Energy Institute
1776 I Street NW
Suite 400
Washington, DC 20006

Mr. Thomas P. Miller

U.S. Department of Energy
Headquarters - Germantown
19901 Germantown Road
Germantown, MD 20874-1290

U. S. NUCLEAR REGULATORY COMMISSION

STAFF ASSESSMENT OF THE APPLICABILITY OF THE AP600

PASSIVE SAFETY SYSTEM ANALYSIS CODES AND TEST PROGRAM

TO THE AP1000 STANDARD PLANT DESIGN

1.0 INTRODUCTION

The AP1000 standard design is a two-loop, pressurized-water reactor (PWR) with an electric output of approximately 1,090 MWe, evolving from the AP600 passive plant design with an electric output of approximately 600 MWe. These advanced plant designs differ from the conventional PWRs in that passive safety systems are used for accident mitigation. Unlike the conventional active safety systems, these passive systems use only natural forces (such as gravity, natural circulation, and compressed gas). These driving forces cool the reactor core following an accident.

Since the AP1000 plant design has a thermal power of approximately 3,400 MWt (compared to 1,933 MWt for the AP600 standard plant design), the major differences from the AP600 design (as summarized in Table 1) are increased capacities of the major components to accommodate the increased thermal output. In particular, the AP1000 reactor system has a taller reactor core with a longer active fuel length, more fuel assemblies, and higher power density; a larger pressurizer; larger steam generators (SGs) with more tubes and larger heat transfer areas; and larger canned reactor coolant pumps (RCPs) with higher head, capacity, and inertia. In addition to a taller containment with a larger free volume, the capability of the AP1000 passive safety systems is increased with a larger core makeup tank (CMT) diameter; a larger in-containment refueling water storage tank (IRWST) and larger injection line diameters; a larger passive residual heat removal (PRHR) system with more tubes, longer tube length, larger inlet and outlet line diameter; and larger valve and flow path diameters for Stage 4 of the automatic depressurization system (ADS-4). Given these differences, Westinghouse Electric Company (Westinghouse) asserted that the AP1000 design represents an incremental change to the AP600 standard plant design, because it maintains and preserves the design configuration and arrangement, key design features, and performance characteristics of the AP600 design. Consequently, Westinghouse concluded that the AP600 test program and the computer codes used for safety analyses of the AP600 design-basis events also apply to the AP1000 design.

During a meeting with Westinghouse on April 27, 2000, the staff of the Nuclear Regulatory Commission (NRC) agreed to a three-phased approach for reviewing the AP1000 design certification application (Phases 1 and 2 constitute the pre-application review, and Phase 3 is the design certification review). The objectives of the Phase 1 review were to identify the review assumptions and issues to be evaluated in the Phase 2 pre-application review. The objective of the Phase 2 review is to resolve those identified issues and identify issues to be resolved in the Phase 3 review. Phase 3 will entail reviewing the design certification application for the AP1000 plant design, including actual safety analyses and other issues.

As part of the Phase 1 review, Westinghouse identified, in a letter dated May 31, 2000 (Reference 1), five fundamental assumptions for the staff to evaluate during the Phase 2 review. The staff communicated the results of its Phase 1 review in a letter to Westinghouse dated July 27, 2000 (Reference 2).

In a letter dated August 28, 2000 (Reference 3), Westinghouse requested that the NRC staff plan for the Phase 2 review to resolve the following items:

- applicability of the AP600 test program to the AP1000 design
- applicability of the AP600 analysis codes to the AP1000 design
- use of additional design acceptance criteria (DAC)
- use of certain AP600 exemptions for the AP1000 design

Subsequently, Westinghouse submitted the following topical reports (References 4, 5, and 6) to support the Phase 2 review of the applicability of the AP600 test program and analysis codes to the AP1000 standard plant design:

- WCAP-15612, “AP1000 Plant Description and Analysis Report,” December 2000
- WCAP-15613, “AP1000 PIRT [Phenomena Identification and Ranking Tables] and Scaling Assessment,” February 2001
- WCAP-15644, “AP1000 Code Applicability Report,” May 2001

WCAP-15612 describes the AP1000 design, comparing it with the AP600 design, and provides a partial, preliminary AP1000 safety analysis and margin assessment. WCAP-15613 presents (1) the AP1000 PIRTs for large-break loss-of-coolant accidents (LBLOCAs), small-break loss-of-coolant accidents (SBLOCAs), long-term cooling (LTC), non-LOCA transients, and the containment response; (2) an overview of the AP600 test program; and (3) scaling assessments of important separate-effects, integral-effects, and containment tests. WCAP-15644 documents the Westinghouse assessment of the safety analysis codes that were developed and approved for the AP600 design certification to determine their applicability for use in the AP1000 design. Specifically, those safety analysis codes are (1) LOFTRAN for non-LOCA transients and SG tube rupture analyses, (2) NOTRUMP for SBLOCA analyses, (3) WCOBRA/TRAC for LBLOCA and LTC analyses, and (4) WGOTHIC for containment analyses.

As part of the AP1000 Phase 2 review, this report provides the staff’s assessment of the application to the AP1000 of the AP600 passive core cooling system test program and the LOFTRAN, NOTRUMP, and WCOBRA/TRAC analysis codes to the AP1000 standard plant design. This assessment is supported with technical assistance from the NRC’s Office of Nuclear Regulatory Research (RES). The assessment of the AP1000 passive containment cooling system and the WGOTHIC code is addressed in Section 3.4.

1.1 Overview of the AP600 Testing Programs and Scaling Assessment

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.47(b)(2)(i)(A) specifies the test program requirements for certification of a standard design that utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions. These requirements essentially mandate that a passive plant vendor must develop and conduct a design certification test program that includes both separate-effects and integral-effects tests. Moreover, those tests must be of sufficient scope to provide adequate data to assess the computer codes used to analyze plant behavior over the range of conditions of normal operation, transients, and accident sequences.

For the AP600 design certification, Westinghouse performed both separate-effects tests and integral-effects tests to investigate the behavior of the AP600 passive core cooling systems (PXS) and to develop a database for validation of the LOFTRAN, NOTRUMP, WCOBRA/TRAC codes used to analyze the design-basis transients and accidents and the WGOTHIC code for containment analyses. In addition, because some of these tests were performed with scaled test facilities, Westinghouse also conducted scaling analyses to demonstrate the acceptability of the test database. The staff's evaluation of the AP600 test programs and scaling assessment, as well as validation of the analysis codes, are documented in Chapter 21 of NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design" (Reference 7).

1.1.1 AP600 Testing Programs

The AP600 passive core cooling tests (summarized in Table 3.1-1 of WCAP-15613) include (1) the separate-effects tests on the passive residual heat removal heat exchanger (PRHRHX), ADS, and core makeup tank (CMT); and (2) the integral-effects tests performed at the Advanced Plant Experiment (APEX) facility and the Simulatore per Esperienze di Sicurezza (SPES) facility.

The PRHRHX separate-effects tests were performed at Westinghouse's Science and Technology Center near Pittsburgh, Pennsylvania, with three vertical tubes submerged in a water tank to determine heat transfer characteristics of the PRHRHX and mixing characteristics in the IRWST. Because the AP600 PRHR has a "C-tube" heat exchanger, Westinghouse provided justifications for the applicability of the straight-tube PRHRHX test data to the "C-tube" configuration. Westinghouse also performed "blind" calculations of selected NRC-provided data, which were obtained from the NRC's confirmatory test program in the Rig of Safety Assessment/Large-Scale Test Facility (ROSA/LSTF) loop in Japan, which employs a simulated C-tube PRHRHX of prototypical bundle dimensions (with approximately one-thirtieth the number of heat exchanger tubes) immersed in a tank of water simulating the IRWST. The results of the Westinghouse calculations of the ROSA facility data demonstrated the adequacy of the straight-tube-based correlation for analysis of the C-tube PRHRHX.

The ADS separate-effects test program at the "VAPORE" facility of the Central Research Establishment of the Italian Energy Agency has a full-size configuration of the piping network, exhaust pipe, and sparger for ADS Stages 1, 2, and 3 (ADS-1/2/3). The tests consisted of two phases. Phase A tests were performed for the ADS-1/2/3 with steam flow through a sparger into a larger water-filled tank to investigate the capacity of the ADS sparger in the IRWST and determine the dynamic effects on the IRWST structure. The Phase B tests were performed to investigate the thermal-hydraulic behavior of the ADS valves, piping, and sparger in the

IRWST. In addition, the tests provided thermal-hydraulic performance data with test conditions that sufficiently covered the operating conditions expected in the AP600 depressurization mode.

Westinghouse did not perform ADS-4 separate-effects testing for the AP600 design. They stated that ADS-4 was sized conservatively and tested as part of the integral-effects tests.

The CMT separate-effects test facility at Westinghouse's Waltz Mill facility in Pennsylvania is a scaled facility that adequately represents the key features of the reactor coolant system (RCS) and connecting piping that could affect CMT performance (This includes the relative elevations of the reactor vessel and CMT, as well as the flow resistance of the pressure balance line and drain line). WCAP-13963 (Reference 8) documented how Westinghouse scaled the AP600 CMT tests. In general, Westinghouse conducted the tests to characterize the CMT over the full range of thermal-hydraulic conditions that the plant will experience. The important phenomena studied included thermal stratification in the CMT, as well as the effects of recirculation, draining, and plant depressurization on CMT behavior. The tests also verified operation of the tank level instrumentation.

The SPES facility is a full-height, full-pressure, scaled full power, integral-effects test facility located at the Societa Informazioni Esperienze Termoidrauliche's (SIET) in Piacenza, Italy. That facility provided data to evaluate the operation of the PXS at high pressure, including response to an SBLOCA, steam generator tube rupture (SGTR), and steamline break transients.

The APEX facility is a ¼-scaled, low-pressure integral-effects test facility at Oregon State University (OSU). That facility provided data to evaluate the operation of the PXS at low pressure in the last part of depressurization and long-term cooling behavior in SBLOCA events. The test matrix focused on SBLOCAs for two reasons. First, within the design basis, LOCAs are the only events that cause the ADS-4 to actuate and to progress to LTC. Second, Westinghouse calculations indicated that the LBLOCA response in the AP600 is similar in many respects to that of conventional designs, and the company asserted that important phenomena related to LTC in an LBLOCA would be similar to SBLOCA behavior. In addition, the Japanese (under contract to the NRC) conducted confirmatory full-height, full-pressure integral-effects tests in the ROSA facility.

1.1.2 AP600 Scaling Assessment

Westinghouse conducted a scaling assessment to demonstrate the acceptability of the test database for the AP600 standard plant design. That assessment was described in Topical Report, WCAP-14727, Revision 2, "AP600 Scaling and PIRT Closure Report" (Reference 9).

The scaling assessment process is founded on the hierarchical, two-tiered scaling (H2TS) analysis methodology, which was first developed by Zuber (Reference 10). The H2TS method consists of a top-down, global, system-level scaling analysis, which considers actively participating systems and components during a transient, and a bottom-up, component-level scaling analysis, which considers important processes that occur locally within a specific region. Building on that foundation, the Westinghouse scaling assessment consists of the following procedures:

- Develop a PIRT for transients and accidents to identify the important phenomena at the component level.

- Perform single-loop, system-level, top-down analyses to identify the thermal-hydraulic processes that are important to the interaction of the components.

- Compare the component-level phenomena in the PIRT with the important system-level thermal-hydraulic processes to ensure that the high-ranked component-level phenomena include those that influence the significant system-level processes.

- Perform bottom-up analyses to calculate the dimensionless time ratios (or Pi groups) of the high-ranked phenomena identified in the PIRT for the SBLOCA and LTC for each component.

- Compare the system-level and component-level Pi groups of the integral-effects tests (APEX and SPES) and separate-effects tests with the AP600 Pi groups to identify distortions in the tests.

The top-down scaling analysis is performed from the system-level conservation equations of mass, momentum, and energy to describe the system response to an accident. The parameters in the system equations are then normalized over their expected range to arrive at dimensionless system equations. The coefficients of the dimensionless system equations are then divided by the expected dominant term. The resulting coefficients for the terms in the normalized equations are the Pi groups, which must be preserved if a test facility is properly scaled.

The values of Pi groups within the normalized equations provide the measures of the relative importance of the parameters. Ratios of the Pi values of the facilities to the plant indicate proper scaling or distortion of the test facilities. If the Pi ratio of the test facility to the AP600 is within the acceptance criteria (between 0.5 and 2), the facility is appropriately scaled for that Pi group.

The Westinghouse scaling analysis considered the regions of the system that were the most active, and the SBLOCA and LTC transients were divided into the following six discrete periods:

(1) single-phase natural circulation with active SGs
(2) single-phase natural circulation with PRHR providing heat removal
(3) two-phase natural circulation with PRHR providing heat removal
(4) ADS blowdown
(5) IRWST injection
(6) sump injection

In the AP600 “single-loop” scaling analysis, energy and momentum equations for a single circulation path were non-dimensionalized, and approximately 40 Pi groups were defined. Scaling ratios for comparisons between the AP600 and SPES and/or APEX facilities were used to determine the scalability of the test data to the full-scale plant.

To augment the single-loop scaling analysis for the AP600 design, Westinghouse also performed a "multi-loop" scaling analysis to ensure that the single-loop approach had not overlooked the effect of a component or subsystem interaction. Specifically, Westinghouse performed the multi-loop analysis for the ADS blowdown and sump injection periods. Those analyses yielded approximately 90 dimensionless groups for the two periods, and the results were consistent with the single-loop analysis. Supporting both top-down scaling approaches was a bottom-up scaling analysis, which included evaluation of approximately 50 dimensionless groups representing processes that occurred at the component level.

The Idaho National Engineering Laboratory (INEL) and the Brookhaven National Laboratory (BNL) assisted the NRC in performing scaling evaluations of the AP600 test program as part of the AP600 design certification review. Those evaluations, documented in References 11 and 12, differed in overall approach and scope, but each attempted to characterize system performance by defining a set of dimensionless Pi groups. The INEL scaling analysis, which focused on determining the Pi groups that impacted the liquid level in the reactor vessel, identified a total of 38 Pi groups, while the BNL's very detailed top-down scaling analysis considered multiple interactions between system components and developed 127 Pi groups.

2.0 SCALING ASSESSMENT OF APPLICABILITY OF THE AP600 TEST PROGRAMS TO THE AP1000 DESIGN

In support of its assertion that the AP600 test program is sufficient to meet the test requirements for a design certification application for the AP1000 standard plant design, Westinghouse submitted Topical Report WCAP-15613 which describes the AP1000 PIRTs and scaling assessment. In addition to evaluating the AP1000 PIRTs and scaling assessment performed by Westinghouse, the staff also performed an independent scaling analysis to address the applicability of the AP600 test program to the AP1000 standard plant design.

2.1 Phenomena Identification and Ranking Tables (PIRT)

Section 2 of WCAP-15613 contains separate PIRTs for LBLOCAs, SBLOCAs, and non-LOCA transients for the AP1000 design, as well as those for the AP600 design. The PIRTs provide a means to identify and classify, in terms of importance, the thermal-hydraulic phenomena expected to occur in transients and accidents that must be included in the analytical models, and for which data must, therefore, be available to evaluate those analytical models. The NRC staff evaluated the AP600 PIRTs during the AP600 design certification review, and found that they capture the important phenomena, processes, and components. The staff also notes that, in general, the AP1000 PIRTs are very similar to those of the AP600 design, with only minor changes (mostly in the importance ranking of certain phenomena). In addition, although Westinghouse developed a separate LTC PIRT for the AP600 design, the company merged the LTC PIRT into the SBLOCA PIRT for the AP1000 design. This is because the sump injection for LTC is also the final phase of the SBLOCA recovery transient. In the combined SBLOCA PIRT, the phenomena that were ranked higher in the LTC PIRT than in the SBLOCA PIRT remain higher. Also, because of the higher steam flow rate expected to result from increased core power in the AP1000 design, hot leg entrainment during the ADS-4 blowdown, IRWST injection, and sump injection phase is increased to a "high" importance ranking from "medium" in the AP600 as is the ADS-4 two-phase pressure drop during the IRWST injection phase. The entrainment/de-entrainment in the upper head and upper plenum region is also increased from “medium” to “high” ranking. In addition, some phenomena ranked as “low” importance for the AP600 design are changed to “medium” importance in the AP1000 PIRT. Westinghouse stated that the AP1000 PIRTs were reviewed by a group of nuclear industry experts, and some of the changes reflect their suggestions.

The staff agrees with the Westinghouse positions that the AP600 and AP1000 PIRTs for LBLOCA, SGTR, and non-LOCA transients are very similar, and no new “high” ranked phenomena are expected. The minor changes in “low” and “medium” ranked processes are considered appropriate.

The staff also finds that the AP1000 SBLOCA PIRT appropriately ranks the important phenomena. The staff requested that Westinghouse evaluate a potential for a condensation-induced water hammer (CIWH) in the direct vessel injection (DVI) line, which could occur when cold CMT or accumulator water contacts a low-velocity, stratified steam-water mixture in the DVI line during the early part of the ADS-1/2/3 blowdown in an SBLOCA. In its response, Westinghouse states that CIWH potential in the AP1000 DVI line is small, as substantiated by the following considerations:

- A comprehensive water hammer assessment performed by Westinghouse for the AP600 design concluded that water hammer potential for the DVI piping is small, given the criteria described in NUREG/CR-6519 (Reference 13) for piping configurations and thermal-hydraulic conditions that can result in water hammer induced by steam bubble collapse.

- Water hammer evaluations for the tests performed at the APEX and SPES facilities found no evidence of significant CIWH events in the DVI lines. There was some evidence of CIWH in the vessel downcomer at the APEX facility during recovery following simulated SBLOCAs when the accumulator injection flow rate was high. However, given the water slug velocity in the downcomer calculated by the RELAP5 code, Westinghouse estimated that the peak pressure in the AP600 reactor is less than 150 psi, which is significantly less than the 400 psid differential pressure for which the affected reactor vessel components are designed.

Because the AP1000 and the AP600 DVI piping and thermal-hydraulic conditions during recovery from design-basis events are almost identical, Westinghouse concluded that the CIWH potential is small, with no need to add the CIWH to the SBLOCA PIRT for the AP1000 standard plant design. The staff also concludes that the CIWH is not a high-ranking phenomenon, given that the DVI line is expected to be full at the initiation of the ADS-1/2/3 blowdown, which would reduce the probability of CIWH at a time when it would be the most severe.

The staff also evaluated a need to increase the ranking of pressurizer phenomena during the ADS blowdown period. Since the AP1000 ADS-1/2/3 valve size is the same as in the AP600 design, velocities during ADS-1/2/3 blowdown are expected to be similar to those in an AP600 plant. The staff concludes that the AP1000 ranking of these processes is acceptable.

In summary, the staff finds that the AP1000 PIRTs (with the changes from the AP600 PIRTs), reflect the changes associated with the AP1000 design characteristics of higher core power density and steam flow rate.

2.2 Separate Effects Tests

Westinghouse performed separate-effects tests for the CMT, ADS valves, and PRHR heat transfer in support of the AP600 system design. Section 4.1.1 of WCAP-15613 describes Westinghouse's evaluation of the applicability of these separate-effects tests to the AP1000 standard plant design. The staff's evaluations of these tests for applicability to the AP1000 design are summarized in the following subsections.

2.2.1 CMT Tests

The AP1000 CMTs have larger diameters and drain rates than do the AP600 CMTs. As described in WCAP-13963 (Reference 8), for scaling of the CMT tests for the AP600 design, key dimensionless Pi group parameters to scale CMT circulation and heat transfer include the Richardson number and the Friction number. In this case, the scaling assessment described in WCAP-15613 shows that the ratio of the Richardson number to the Friction number for the AP1000 CMT remains acceptably close to those in the test matrix. In response to the staff's Request for Additional Information (RAI) P47, Westinghouse showed that other CMT scaling groups (e.g., the Stanton number, the liquid heat source ratio, and the heat source ratio) that are affected by the larger CMT diameter and drain rate also remain reasonably scaled for the AP1000. Therefore, the CMT tests can be considered to be acceptable for the AP1000 design as they were for the AP600 design. The staff therefore concludes that the CMT tests remain valid for the AP1000 code validation.

2.2.2. ADS-1/2/3 Valve Tests

The ADS-1/2/3 system for the AP1000 standard plant design is identical to that of the AP600 design. Since the flow through the ADS-1/2/3 is expected to be choked during its blowdown, and since simulations have revealed that upstream pressures in the AP1000 are very similar to those in the AP600 design, thermal-hydraulic conditions affecting ADS-1/2/3 performance will be close to those in the AP600 design. Consequently, tests performed to investigate ADS-1/2/3 valve performance for the AP600 design, which included a wide range of actuation pressures and flow qualities, are considered appropriate to represent conditions in the AP1000 standard plant design.

Westinghouse did not perform ADS-4 separate-effects testing for the AP600 design. The company stated that ADS-4 was treated/sized conservatively and tested as part of the integral-effects tests, and Westinghouse will take the same approach for the AP1000 standard plant design. Sections 2.3.2 and 2.4 of this report discuss the staff's evaluation of the ADS-4 testing.

2.2.3 PRHR Heat Transfer Tests

The AP1000 PRHR maintains the same "C-tube" heat exchanger, tube diameter, spacing, and pitch ratio of the AP600 PRHR. To accommodate the higher core power, however, the

AP1000 PRHR line resistances are reduced to increase the natural circulation flow, and the PRHRHX heat transfer area is increased by about 22 percent by adding tubes to the top and bottom of the tube sheet and adding length to the horizontal sections. Thus, a higher proportion of the heat transfer is expected to occur by crossflow through the horizontal section in the AP1000 design than in the AP600 design.

The PRHR tests for the AP600 standard plant design, which were performed with straight vertical tubes, did not include results for horizontal crossflow. However, Westinghouse conducted a subsequent analysis using data from the ROSA facility using a scaled C-tube heat exchanger to show that the heat transfer correlations developed from the vertical tube data were acceptable for heat transfer performance of the PRHR C-tube heat exchanger.

During the AP600 review, a concern was raised regarding the potential for a drastic reduction in heat transfer caused by vapor blanketing attributable to violent boiling on the outer tube surface in the top horizontal tube region of the PRHR heat exchanger. This concern was resolved on the basis of Westinghouse's analyses of the margin of the PRHR heat exchanger heat flux to the critical heat flux limit, and the fact that vapor blanketing was not observed in the APEX, SPES, and ROSA integral-effects test facilities. Section 4.1.1.3.2 of WCAP-15613 provides an evaluation of the AP1000 PRHR, using various heat transfer correlations on the inside and outside of the PRHR tubes to determine the heat flux on the outside of the tubes and margin to the critical heat flux limit. The results show that the expected operating conditions for the AP1000 PRHR result in external tube heat flux values that are far below the critical heat flux limits, and are bracketed by forced flow test data from the AP600 integral-effects tests. Westinghouse therefore concludes that heat transfer correlations that were developed from the AP600 test data remain valid for the AP1000 PRHR.

The staff's evaluation of the effect of increasing the horizontal length on the overall heat transfer has concluded that no additional testing at the Westinghouse separate-effects test facility is required. Acceptability of the Westinghouse codes to predict PRHR heat transfer depends on the simulation of the integral-effects data from the ROSA facility.

2.3 Scaling Analysis of Integral Effects Tests

The integral-effects tests from the APEX, SPES and ROSA facilities provide experimental data for the AP600 code validation. However, Westinghouse simulated and used only the SPES and APEX facilities for the AP600 code validation and design certification. The applicability of the AP600 integral-effects tests to the AP1000 code validation and design certification is evaluated through both top-down and bottom-up scaling analyses, as described in the following subsections.

2.3.1 Westinghouse Scaling Analysis

Section 4 of WCAP-15613 documents Westinghouse's scaling evaluation to demonstrate the applicability of the AP600 test program database to the AP1000 safety analysis code validation. Specifically, that scaling evaluation provides a quantitative means to show how well important phenomena are preserved in the test facilities that were originally scaled for the AP600 plant design relative to the AP1000 design.

As in the AP600 scaling evaluation of the passive core cooling system test facilities, the top-down system-level scaling analysis of the integral-effects tests is based on the SBLOCA transients. This is because an SBLOCA transient includes broad ranges of thermal-hydraulic behavior, and all of the PXS safety features are employed during the transient.

In an SBLOCA, the reactor coolant system (RCS) depressurizes during initial blowdown through the break. As the safeguard ("S") signal actuates the passive safety system, the RCPs trip quickly, and the RCS passes into natural circulation. In the early stage, the RCS experiences single-phase natural circulation, with the SGs providing the dominant heat sink. This is followed by a later phase when the PRHR becomes the dominant heat sink after the SGs have drained. As the primary system drains, it passes into two-phase natural circulation, in which a mixture exists in the cold and hot legs; the CMT cold leg pressure balance line is either two-phase or steam, and the CMTs are draining. There is boiling in the core and a two-phase mixture leaves the core and flows into the hot legs. Steam or a two-phase mixture enters the PRHR with single-phase water leaving.

A similar behavior occurs in the CMTs, in which a two-phase mixture or steam enters the cold leg balance line and liquid flows from the CMT to the vessel in the DVI line. As the CMT drains to a level of 67.5 percent, ADS-1 is actuated, followed by ADS-2/3, resulting in RCS depressurization by venting the steam from the pressurizer to the IRWST. The accumulators also inject borated water into the RCS as it depressurizes below the accumulator pressure. During the ADS-1/2/3 blowdown phase, a portion of the system (such as the DVI line, vessel downcomer, and lower plenum) remains single-phase. The remainder of the system is two-phase, including the core, upper plenum, hot legs, pressurizer, and pressurizer surge line, which now fills in response to the activation of ADS-1/2/3. As the CMT drains to 20 percent, ADS-4 is actuated, and its blowdown further depressurizes the RCS to enable IRWST injection. The ADS-4 blowdown transition to the inception of IRWST injection is considered critical in the AP1000 passive plant design because it is in this period that minimum inventory in the reactor vessel is expected to occur. During the IRWST injection, the RCS is an open system with the IRWST feeding the reactor vessel by gravity injection, which flows through the DVI line into the downcomer, then up and around the downcomer and out the break to the sump, or down the downcomer into the core and out the ADS-4 valves on the hot legs to the sump or containment. As the IRWST drains, sump injection is initiated. The sump injection period is similar to the IRWST injection, with the exception that the system is now a closed loop with the primary system coupled to the containment, which provides for LTC.

Westinghouse divided the SBLOCA transient into the following six phases:

- initial blowdown
- natural circulation
- ADS-1/2/3 depressurization
- ADS-4 to IRWST transition
- IRWST injection
- sump injection

One major difference in this breakdown of the transient phases from that of the AP600 design is the addition of the ADS-4 to IRWST transition, which is the most important phase in an SBLOCA as the minimum mixture level in the reactor vessel is expected to occur during this period.

As in the AP600 scaling analysis, the Westinghouse AP1000 scaling analysis does not consider the initial blowdown phase because it is a relatively short period common to both current operating plants and the advanced passive plant design and does not involve passive safety system components.

For the top-down scaling analysis, system-level conservation equations are written to address the important processes and parameters that are involved in each specific phase. The equations are combined in a form which identifies the physical processes and key parameters of interest, such as reactor vessel inventory, pressure, quality, or void fraction. The variables in the combined equations are non-dimensionalized using reference values appropriate for the specific period of the transient, and the resulting dimensional coefficients in the equations are then normalized using the coefficient of the dominant process. The end result yields dimensionless Pi groups. The test facility/plant scaling ratios of these Pi groups are then calculated and compared to the acceptance criteria to determine if the test facility is sufficiently scaled to the full-scale plant.

For the natural circulation phase, two-phase natural circulation with PRHR providing heat removal is analyzed by combining the steady-state mass, momentum, and energy equations into a core exit quality scaling ratio expression in terms of the dominant influences (such as PRHR gravity head, PRHR flow path hydraulic resistance, and core decay power). The scaling ratios of core exit quality between the test facility and the AP1000 indicate that the SPES facility is sufficiently scaled for both the AP600 and the AP1000, whereas the APEX facility is not well-scaled for the natural circulation phase.

For the ADS-1/2/3 blowdown depressurization phase, the scaling analysis is performed with the rate of pressure change equation for the ADS depressurization process. The analysis produces the scaling ratios of two Pi groups; one group is the ratio of core steam generated by the decay heat to RCS steam volume, and the other is the ratio of the steam venting through ADS-1/2/3 to the RCS steam volume. The resulting scaling ratios show that the SPES facility is sufficiently scaled to the AP600 and AP1000 designs; however, the APEX facility has distortion in the ratio of ADS-1/2/3 steam venting to the RCS steam volume.

The top-down scaling analysis of the ADS-4 to IRWST transition phase considers the CMT-injection dominating subphase, the IRWST-injection dominating subphase, and the ADS-4 depressurization phase. For the CMT-injection and the IRWST-injection dominating subphases, the scaling analyses are derived from the transient equations of the reactor vessel inventory. For the ADS-4 blowdown, the scaling analysis is derived from the rate of RCS pressure change. The scaling analyses of these subphases generate seven Pi groups. The facility/plant scaling ratios of these Pi groups showed that the APEX and SPES facilities are sufficiently scaled to both the AP600 and the AP1000 designs, when the ADS-4 flow is critical. When the ADS-4 flow is subcritical, the SPES facility is distorted as a result of the oversized ADS-4 vent paths.

Westinghouse also performed a scaling analysis on the basis of the NRC-sponsored tests at the APEX facility for the AP600 design. These tests included a series of 10 “core uncovery tests,” in which the RCS was drained to the hot leg level and the IRWST was pressurized to simulate AP600 IRWST gravity injection. The ADS-4 vents were used to depressurize the system. The top-down analysis of the IRWST injection, where the two-phase resistance dominates, derived a Pi group for the equilibrium quality. The scaling ratio of this Pi group indicates that the APEX facility is sufficiently scaled for the AP1000 standard plant design.

The scaling analyses of the IRWST and sump injection phases are performed to determine the core exit quality, which impacts the thermodynamic state, two-phase flow regime, and pressure drop. By combining the conservation of mass, momentum, and energy, an expression is developed for the core exit quality. Expressions are then derived for the core exit quality scaling ratio, which contain a density ratio, a gravity head to resistance ratio, and a core power to enthalpy ratio. The scaling ratios show that the core exit quality of the APEX facility is sufficiently scaled to the AP600 and the AP1000 designs; however, the SPES facility is not well-scaled to either design, as it did not simulate sump injection.

It should be noted that the top-down scaling approach used in the AP1000 review is not the same approach used in licensing the AP600 design, as documented in WCAP-14727. Unlike the AP600 design, the AP1000 top-down scaling approach combines the mass, momentum and energy equations into a single expression for the parameter of interest and significantly reduces the number of scaling groups.

To complement Westinghouse's top-down scaling analysis for the AP1000 standard plant design, the staff requested (in RAIs P55 through P58) that Westinghouse provide the AP1000 numerical values of those Pi groups listed for the AP600 design in Tables 3.2-8 through 3.2-12 of WCAP-14727, which were derived directly from the separate momentum and energy equations. Assessments of these Pi groups would provide consistency with that accepted for the AP600 design. In response to the staff's RAIs, Westinghouse provided the numerical values of these Pi groups for the two-phase natural circulation, ADS blowdown, and IRWST and sump injection phases. For each Pi group in a transient phase, Westinghouse provided the Pi value for the SPES and APEX facilities and the AP600 and AP1000 plants, as well as the facility-to-AP1000 scaling ratios.

For the majority of the Pi groups, the scaling ratios between the facility and the AP1000 design are within the acceptance criteria. For certain Pi groups, the scaling ratios are outside of the acceptance criteria, indicating scaling distortion; however, some of the distorted Pi groups are insignificant as indicated by their small Pi values relative to other more dominating terms. These insignificant Pi groups include the inertia-to-buoyancy ratio, the phase change momentum flux-to-buoyancy ratio for the natural circulation phase, the single-phase pressure compliance-to-core power ratio, the two-phase mechanical compliance-to-core power ratio for the ADS blowdown phase, and the inertia and momentum flux terms for the IRWST and sump-injection-phases.

For the IRWST injection phase, the scaling ratios of the Pi group of the resistance-to-buoyancy ratio for the SPES and APEX facilities are outside of the acceptance criteria. Westinghouse states that the formulation of this scaling group was derived from the AP600 program on the basis of the single-phase contribution to resistance of the DVI and ADS paths, which significantly understates the two-phase resistance associated with the ADS flow path. Therefore, the scaling of this phase was reformulated for the AP1000 to account for the two-phase resistance associated with the ADS flow path in WCAP-15613. The results show that the APEX facility is well-scaled to the AP1000 design, while the SPES facility shows distorted scaling.

In the ADS blowdown phase, the scaling ratios show that the SPES facility has distortions for the Pi groups of the boiling heat to core power ratio and the single-phase mechanical compliance-to-core power ratio, while the APEX facility has distortion in the sensible heat-to-core power Pi group.

To supplement the top-down system-level scaling analyses, the bottom-up scaling analyses are performed for the important local processes or phenomena (during various phases of the transient) that are not captured in the top-down scaling analysis.

For the natural circulation phase, the bottom-up scaling analyses are performed for the flow patterns and phase separation at the cold leg T-junction at the CMT balance line (CMTBL). The cold-leg flow pattern is analyzed on the basis of the Taitel-Dukler horizontal flow regime transition map (Reference 14), and the facility/plant scaling ratio of the Froude number is calculated. The resulting scaling ratios show that the APEX and SPES facilities are sufficiently scaled to both the AP600 and the AP1000 design. The scaling analysis of phase separation at the cold leg-CMTBL junction is performed on the basis of the correlation developed by Seeger, et al. (Reference 15), for a top vertical branch in a non-stratified upstream flow regime, which correlated the quality ratio to the mass flux ratio of the branch and the main pipes. The facility/plant scaling ratio of the balance line-cold leg quality ratio showed that the APEX and SPES facilities are sufficiently scaled to the AP600 and the AP1000 designs.

During the initial stage of the ADS-1/2/3 blowdown when only steam is vented, Westinghouse states that the APEX facility surge line length-to-diameter ratio and surge line layout are preserved relative to the AP600 standard plant design to preserve the surge line pressure drop. Because those values are unchanged in the AP1000 design, the surge line pressure drop should also be preserved for that design. However, in the later stages of ADS-1/2/3 depressurization, a two-phase mixture flows through the surge line into the pressurizer. Westinghouse states that Reyes found that the APEX facility was probably distorted for some flow patterns (such as the slug-annular flow regime transition) with respect to the AP600 design. Therefore, it is expected that some distortion may also exist with respect to the AP1000 design. In addition, the SPES facility length-to-diameter ratio is not scaled, and the flow pattern transition scaling analysis was not performed. Westinghouse contends that although there may be some distortion of flow regime in the surge line of the SPES and APEX facilities, it should only affect the later stages of the ADS depressurization when a two-phase mixture is discharged. However, since venting of the gas phase has a higher impact on RCS pressure than the discharge of the liquid phase, the surge line pressure drop should be acceptably scaled for the steam venting regime and, therefore, the data from the test facilities can be used during the ADS phase for code validation.

For the ADS-4 to IRWST transition phase, the bottom-up scaling analysis considered hot leg flow pattern, liquid entrainment from hot leg into the ADS-4, and counter-current flow in the surge line during pressurizer draining.

Like the cold leg flow pattern, the hot leg flow regime transition from stratified to non-stratified flow is an important phenomenon as it influences pressure drop and entrainment in the ADS-4 flow path. Taitel-Dukler's general flow regime map for the horizontal two-phase flow is used to predict flow regime transitions. A scaling ratio expression between the test facility and plant is derived on the basis of preserving the modified Froude number used in the Taitel-Dukler flow regime map and pressure similitude. The scaling ratios show that the APEX and SPES facilities are sufficiently scaled to the AP1000 design.

The scaling analysis for ADS-4 entrainment is performed for the onset of liquid entrainment on the basis of the following correlation of the onset of liquid entrainment for a vertical offtake with stratified flow in the main pipe:

$$Fr = [J_g / (gd)^{0.5}] (\rho_g / \Delta\rho)^{0.5} = 5.7(h_b / d)^{1.5}$$

Where Fr is the Froude number, J_g is the superficial velocity of steam, g is gravitational acceleration, ρ_g is the density of steam, $\Delta\rho$ is density difference, d is the off-take pipe diameter, and h_b is the distance from the top of the pipe to the stratified level.

A scaling ratio relation for the entrainment onset is derived assuming pressure similitude. The APEX/AP1000 scaling ratio entrainment onset is calculated to be 0.69, which indicates that the APEX facility is sufficiently scaled; however, the SPES facility has distortion with a scaling ratio of 0.14.

The scaling analysis for the countercurrent flow in the surge line and pressurizer draining is performed on the basis of the Kutateladze flooding relation. Westinghouse examined the scaling of the Kutateladze number during this transition phase with the pressurizer draining. For scaling purposes, because the pressurizer is poorly vented as the ADS-1/2/3 path is plugged by a column of water above the sparger in the IRWST during this phase of a transient, the mode of pressurizer draining can be described as an equal volume replacement process so that the superficial velocities of liquid and steam in the Kutateladze number are equal. With this assumption, the scaling relationship simply states that the superficial velocity (and hence the Kutateladze number) is preserved in the test facilities and plants as pressure similitude exists.

For the IRWST injection phase, the bottom-up scaling analysis is performed for the reactor core void fraction on the basis of the Yeh correlation (Reference 16). By preserving the void fraction between the test facility and the plant, the scaling ratio of the Pi group for the core exit void fraction is derived. The scaling ratio of the Pi group shows that the APEX facility is sufficiently scaled.

2.3.2 NRC's Independent Scaling Assessment

The NRC Office of Nuclear Regulatory Research (RES) performed an independent scaling assessment to determine whether the AP600 test program also applies to the AP1000 standard plant design. The staff review and assessment of scaling did not address containment phenomena and was limited to those affecting the AP1000 primary system. The review and assessment of the applicability of test programs to the AP1000 design considered both the primary and containment systems. Severe accident thermal-hydraulics were not part of the Phase 2 review, and will be addressed in Phase 3.

During its assessment, RES performed both top-down system-level and bottom-up process scaling evaluations of the SPES, ROSA, and APEX facilities for applicability to the AP1000 code validation and confirmation of safety margin. In general, at least one facility is well-scaled for the AP1000 standard plant design during the early, high-pressure blowdown periods, and later after sump injection occurs. However, the transition from ADS-1/2/3 blowdown to IRWST injection shows distortions that raise significant concerns. The NRC scaling evaluation follows the methodology developed by INEL (Reference 11) to evaluate scaling for the AP600 standard plant design.

The independent scaling analysis considered five separate periods:

- subcooled blowdown
- intermediate (ADS-1/2/3 venting)
- ADS-4 blowdown
- IRWST injection
- sump injection

The intermediate and IRWST injection periods were also divided into subphases to examine additional system processes.

The following paragraphs discuss the staff's conclusions regarding these scaling evaluations beginning with a summary of the top-down scaling analysis.

<u>Subcooled Blowdown Phase</u>

The subcooled blowdown phase is initiated by the break, and ends just after the pressurizer drains. Differences in core power and pressurizer volume between the AP1000 and the AP600 designs affect some scaling groups. However, no significant distortions were found by comparing the AP1000 Pi groups to those of the SPES and ROSA facilities. Therefore, code validation on the basis of the SPES facility is acceptable.

<u>Intermediate (ADS-1/2/3) Blowdown Phase</u>

The intermediate blowdown phase is considered to be composed of three subphases. Subphase I begins with pressurizer draining and extends to when the hot legs, upper head, and SG reach saturation pressure. Subphase II extends from the end of Subphase I to the initiation of net inflows to the RCS from the accumulators or CMTs. Subphase III extends from the initiation of accumulator injection or CMT draining to the opening of ADS-4. During the intermediate periods, the ADS-1/2/3 system actuates, the PRHR becomes active, and the CMTs begin to drain, as follows:

- Intermediate Subphase I: During this period the most important Pi group for the AP1000 design was found to have better agreement with the SPES facility than those for the AP600 design (There may, however, be some distortion in comparisons of Pi groups with minor importance). Code validation on the basis of the SPES facility data is therefore acceptable.

- Intermediate Subphase II: In general, scaling groups for this period were found to have good agreement between the SPES facility and both AP1000 and AP600 designs. No significant, non-conservative distortions exist and, thus, the SPES facility is adequate for code validation. With regard to PRHR performance, the AP1000 design exhibited better agreement with the ROSA Pi groups. Thus, conclusions regarding simulation of PRHR heat transfer have higher confidence if based on the ROSA facility rather than the SPES facility. However, overall code validation on the basis of the SPES facility data is considered acceptable.

- Intermediate Subphase III: The Pi groups for this period show good agreement between the the SPES facility and both AP1000 and AP600 designs. Differences between scaling groups for AP1000 and the SPES facility are either small or conservative. Therefore, code validation on the basis of the SPES facility is acceptable.

<u>ADS-4 Blowdown Phase</u>

The staff's top-down scaling analysis shows that there may be distortions during the ADS-4 blowdown period. Early in this period when the system pressure is high, the flow is critical. Assuming critical flow, the SPES and ROSA facilities are appropriately scaled for the AP1000 standard plant design conditions during ADS-4 blowdown. The APEX facility however, was found to have non-conservative distortions. The analysis considered a 1-inch cold leg break and a double-ended DVI break, and found that the SPES facility is appropriate, in both scenarios but the APEX facility is not appropriate (By contrast, for the AP600 standard plant design, this approach found that the APEX facility is acceptable, but the SPES facility has conservative distortions). Eventually, the system pressure decreases and the ADS-4 flow becomes non-critical. Assuming non-critical flow from the system, the APEX facility becomes appropriately scaled for the AP1000 design based on scaling groups defined in the INEL scaling methodology.

On that basis, code validation using the SPES facility is considered acceptable during the high-pressure phase of the ADS-4 blowdown, but the APEX facility is not considered acceptable until late in the period when the IRWST transition is about to occur.

This conclusion conflicts with the Westinghouse scaling analysis and the conclusion that the APEX facility is appropriately scaled while flow is critical in the ADS-4 to IRWST transition period. The scaling methodology in Reference 5 defines dimensionless groups and calculates values showing that the SPES and APEX facilities are correctly scaled. The response to RAI P56, however, lists "single-loop" scaling groups for the ADS blowdown phase, and Westinghouse cited both the SPES and APEX facilities as having distortions, yet are considered by Westinghouse to be acceptable for code validation. The staff concludes, however, that code accuracy and validation in the ADS-4 transition period should be based on the SPES facility simulation.

<u>IRWST Injection/Drain Phase</u>

Results of the top-down scaling analysis show that the APEX facility is appropriately scaled for the AP1000 standard plant design. Code validation on the basis of that facility is therefore acceptable.

IRWST/Sump Injection Phase

Results of the top-down scaling analysis show that the APEX facility is appropriately scaled for the AP1000 standard plant design. Code validation on the basis of that facility is therefore acceptable.

Bottom-Up Scaling Analysis

The bottom-up scaling analysis conducted is as follows:

- Froude number comparisons indicate that the SPES facility appropriately scales both the hot and cold leg flow regimes for the high-pressure periods, and the APEX facility appropriately scales these regimes for the low-pressure periods of the SBLOCA and LTC.

- The experimental data in the integral-effects tests is not considered sufficient to validate code models for entrainment and carryover for the AP1000 standard plant design.

- Entrainment in the hot leg and carryover into the branch lines leading to the ADS will occur to a greater extent in the AP1000 design than in the AP600 design or the test facilities. Westinghouse has not yet demonstrated that the existing data is sufficient to validate hot leg entrainment models for the AP1000 design because the company is basing its scaling evaluation on a correlation that may not be applicable to the AP1000 geometry. Specifically, the AP1000 hot leg-to-branch line diameter ratio is significantly different than the ratio used in developing the entrainment onset correlation. Alternative evaluation and scaling of entrainment onset leads to the conclusion that entrainment is more prevalent and will occur at lower hot leg water levels in the AP1000 design than in the tests.

- None of the integral-effects tests appropriately scale the facilities entrainment from the pool of water in the upper plenum above the upper core plate for the AP1000 standard plant design, and Westinghouse did not consider this process in its bottom-up scaling evaluation. The staff's evaluation of entrainment from the upper plenum pool shows that the rate of entrainment in the AP1000 design will be significantly higher than shown in the integral-effects tests.

2.4 Test Program Scaling Assessment Findings

Given the evaluation discussed above, the staff finds that the AP600 test program is generally applicable for code validation of the AP1000 standard plant design. However, the staff also finds that additional validation is necessary for the liquid entrainment phenomena.

The ADS-4 blowdown period to the inception of IRWST injection is critical in the AP1000 passive plant design because it is during this period that minimum inventory in the reactor vessel is expected to occur. Compared to the AP600 standard plant design, the AP1000 design has 75 percent higher core power and (therefore) higher steam flow in the upper plenum, hot leg, and ADS discharge during the ADS-4 blowdown. Even though the AP1000 hot leg diameter remains the same (31 inches) as in the AP600 design, the diameters of the ADS-4 valves and the off-take pipe from the hot leg are increased from 10 and 12 inches to 14 and 18 inches, respectively. The higher steam flow and larger ADS-4 diameter will affect liquid entrainment through the ADS-4 discharge.

As described in Section 2.3.2 above, the NRC staff's top-down scaling analysis revealed that, during the early phase of the ADS-4 blowdown when the flow is critical, the APEX facility has a non-conservative distortion. The staff therefore requested that Westinghouse justify the basis for the acceptability of the AP600 code validation for the AP1000 design, or determine whether additional AP1000 testing is necessary for code validation of the ADS-4 blowdown. In its response, Westinghouse stated that it does not agree with the staff's conclusion that the APEX facility is not suitably scaled for the ADS-4 blowdown phase, and additional hot leg entrainment data for ADS-4 blowdown is not needed for AP1000 code validation. Even though it is expected that higher liquid entrainment may occur in the AP1000 design than in the AP600 design during the ADS-4 blowdown, Westinghouse contends that this does not render the AP600 code validation unacceptable for the AP1000 during the ADS-IRWST transition phase. Moreover, Westinghouse contends that the APEX test facility showed significant entrainment during the ADS-4 blowdown phase. The staff does not agree with this finding.

Westinghouse's scaling assessment for the ADS-IRWST transition phase includes both top-down and bottom-up analyses. The overall top-down scaling analysis generates several Pi groups. The facility/plant scaling ratios of these Pi groups show that the APEX and SPES facilities are sufficiently scaled to the AP600 and the AP1000 designs for choked ADS-4 flow and with respect to core power. When ADS-4 flow is subsonic, the SPES facility is distorted as a result of its oversized ADS-4 vent paths.

Although Westinghouse's bottom-up scaling analysis of entrainment onset showed that the APEX facility is well-scaled for the AP1000 standard plant design, the staff finds that this analysis contains several shortcomings.

- Westinghouse's scaling analysis is based on an entrainment onset correlation in which the applicability to the AP1000 geometry has not been confirmed. This correlation is founded on experimental data with a small branch line to main pipe diameter ratio (d/D), which may not be appropriate for the AP1000 design because it has a large d/D ratio.

- Existing correlations are based on tests performed with small offtake diameter more than 10 times smaller than the main pipe diameter, as summarized by Ardron and Bryce (Reference 17). In the AP1000 design, the ADS-4 branch pipe diameter is 14 inches relative the hot leg diameter of 31 inches, yielding a (d/D) ratio much larger than the AP600 test data.

- The general entrainment onset correlation does not account for the effect of viscosity and liquid surface tension, which may affect the liquid entrainment. Correlations that account for these parameters suggest that significant entrainment will occur for the AP1000 design, but will not occur in the tests Westinghouse has used for code validation.

The staff also finds upper plenum pool entrainment to be an issue for the AP1000 design.

Experiments in the APEX facility as well as in simulations of the AP600 design showed that the double-ended guillotine break (DEGB) of one of the DVI lines and a 10-inch cold leg break could lead to the minimum vessel inventory or core uncovery. Entrainment of liquid from the upper plenum will be significant, and will be more important in the AP1000 design than in the AP600 design. RES considered bottom-up scaling of upper plenum entrainment. Pool entrainment is a complex process that is highly dependent on the gas velocity bubbling through the pool, and the height to which droplets and other entrained liquid must be elevated to exit the vessel.

The AP1000 core power is 75 percent higher than in the AP600 design, but the upper plenum design is nearly identical. The entrainment is often defined as the ratio of the droplet upward mass flux to the gas mass flux:

$$E_{fg} = \rho_f J_{fe} /(\rho_g J_g)$$

Where ρ_f and ρ_g are liquid and gas phase densities, J_g is the gas superficial velocity, and J_{fe} is the entrained phase superficial velocity.

Expressions for E_{fg} show the functional dependence:

$$E_{fg} \propto (J_g)^n$$

The exponent n is generally 3 or higher. Assuming pressure similitude and preserving the dimensionless height ratio, the AP1000 upper plenum pool entrainment can be expected to be at least 1.75^3 or 5.36 times as large as that in the AP600 design. Consideration of experimental tests scaled to the AP600 design power levels leads to the conclusion that the AP1000 upper plenum entrainment is significantly higher than entrainment in the integral-effects tests.

The staff finds that none of the integral-effects test facilities are sufficiently well-scaled so that they provide an acceptable database to validate thermal-hydraulic codes for the high rates of liquid entrainment that are expected to occur in the AP1000 design during ADS-4 and IRWST injection periods of an SBLOCA. Westinghouse has not demonstrated that the existing AP600 integral-effects tests provide data over the range of conditions necessary to validate entrainment models in the codes that the company intends to use. The staff concludes that Westinghouse must either obtain entrainment test data applicable to the AP1000 steam flow rates for code validation, or provide proper justification for the entrainment models to be used for the AP1000 applications.

3.0 ASSESSMENT OF APPLICABILITY OF THE AP600 ANALYSIS CODES TO THE AP1000

For the AP600 design certification, the safety analysis of the design-basis transients and accidents were performed with the following computer codes:

- LOFTRAN for non-LOCA transients and SGTR analyses
- NOTRUMP for SBLOCA analyses
- WCOBRA/TRAC for LBLOCA and LTC analyses
- WGOTHIC for containment analyses

These safety analysis codes were validated with the test data from the AP600 test program, and WCAP-15644 documents Westinghouse's assessment of these safety analyses to determine their applicability and use for the AP1000 design certification. Sections 3.1 through 3.4 present the staff's assessment of the applicability of these codes to the AP1000 standard plant design.

3.1 LOFTRAN Code Applicability

The LOFTRAN code simulates a multi-loop reactor system by modeling the reactor core and vessel, hot and cold leg piping, SG tube and shell sides, pressurizer, and RCPs, with up to four reactor coolant loops. The pressurizer model includes the effects of pressurizer heaters, as well as the operation of spray, relief, and safety valves. The reactor core model employs a lumped fuel heat transfer model with point neutron kinetics, and includes the reactivity effects of variations in moderator density, fuel temperature (Doppler), boron concentration, and control rod insertion and withdrawal. The secondary side of the model uses a homogenous, saturated mixture for thermal transients and a water level correlation for indication and control. The code also models the safety injection system, including the accumulators and the effects of pump coastdown and pump startup. Flow reversal in the reactor coolant loops is allowed, except in the loop with the pressurizer where flow reversal is not allowed.

The LOFTRAN thermal-hydraulic model is best suited for use in transients in which the primary coolant system remains subcooled. The model may also be used for a main steamline break analyses, where two-phase conditions occur in the upper reactor vessel head. The upper head is a hydraulically stagnant region, which receives only a small fraction of the main coolant flow. For accident conditions in which the extent of voiding extends beyond the pressurizer and the upper head, use of the LOFTRAN code would not be appropriate without additional justification.

The LOFTRAN code does not have a detailed core heat transfer model. An overall fuel rod-to-coolant heat transfer coefficient (UA) is utilized which is a parabolic fit to values specified by the user. Input values either maximize or minimize core heat transfer, depending on the conservative direction for the transient of interest. The inputs are obtained from the limiting values predicted using detailed Westinghouse fuel rod design codes. For evaluations in which accurate knowledge of core heat transfer or fuel temperature is important, physical conditions are transferred from LOFTRAN to more detailed thermal-hydraulic codes, such as THINC, FACTRAN, and VIPRE.

The NRC staff found (Reference 19) that the LOFTRAN code is acceptable for analysis of transients and accidents at operating plants, as presented in Chapter 15 of the plant safety analysis reports. This approval did not extend to transients involving a LOCA or SGTR.

In order to model an SGTR, Westinghouse modified the LOFTRAN code to include an enhanced SG secondary-side model, a tube rupture break flow model, and improvements to allow simulation of operator actions. This modified version of the code, which is sometimes referred to as LOFTTR2, was reviewed and approved by the NRC staff, as discussed in References 20, 21, and 22.

3.1.1 Application of LOFTRAN to Passive Plants

Westinghouse made additional modifications to the LOFTRAN code to model the AP600 standard plant design, as described in Westinghouse topical reports WCAP-14234 and WCAP-14307 (References 23 and 24). Table 2 lists the transients and accidents for which LOFTRAN has been approved, given the consideration of possible failure of the AP600 passive safeguard systems. This approval is discussed in the NRC staff's safety evaluation report (SER) for the AP600 standard plant design (Reference 7).

References 4 and 6 describe the use of the LOFTRAN code for the AP1000 evaluations. Reference 4 contains a general description of the AP1000 standard plant design and preliminary analyses of a subset of the transients and accidents listed in Table 2. The subset was selected by Westinghouse to illustrate performance of the AP1000 passive safety features and plant differences between the AP600 and the AP1000 designs. Reference 6 presented additional details and justifications for use of the LOFTRAN code for AP1000 analyses. Reference 5 gives a revised PIRT for the AP1000 transients and accidents, with a scaling assessment of the tests used to qualify the LOFTRAN code.

Since an AP1000 plant will be similar in design to an AP600 plant, Westinghouse believes that modifications made to the LOFTRAN code to model the AP600 standard plant design will also address the AP1000 design. The size differences between the two designs can be accounted for in the code inputs. The fuel, pressurizer, and SGs for the two passive plant designs will be similar to those used in operating plants. Unlike operating plants where the hot and cold leg nozzles are at the same elevation on the reactor vessel, the hot and cold leg nozzles for the AP600 and the AP1000 designs are at different elevations. The elevation difference is accounted for by LOFTRAN modifications. The RCPs for the passive plants are of a canned rotor design with their own characteristics for developed head and torque as functions of flow rate and impeller speed. The mass of the pump flywheel was increased for the AP1000 standard plant design to provide for a longer flow coastdown in the event that an RCP should inadvertently trip during operation. The pump characteristics are accounted for by inputting the proper information from the pump manufacturer. Since full pump characteristics can be input into LOFTRAN, the code should be able to model the RCPs of the AP1000 design when properly described in the input.

Operating Westinghouse plants use a single cold leg and RCP per coolant loop, while the AP600 and the AP1000 designs use two cold legs and two RCPs per loop. The LOFTRAN code is capable of evaluating the dual cold leg loop arrangement, provided the two cold legs in a loop have the same behavior (so that they can be lumped together). When the two cold legs do not have the same behavior, such as a tripped RCP or locked rotor/sheared RCP shaft, Westinghouse inputs the net cold leg flow rate as a function of time (that rate is calculated using external methods). Westinghouse presented the asymmetrical cold leg methodology to the NRC staff for review as part of the review of the LOFTRAN code for the AP600 design review. In that instance, the NRC staff concluded that the external flow calculational methodology is acceptable. This same methodology of calculating the asymmetric cold leg flow rates outside of the LOFTRAN code remains acceptable and will be used for the AP1000 design analyses.

Westinghouse also modified the LOFTRAN code by adding the capability to model the following additional components, which are part of the AP600 and AP1000 designs but are not present in operating plants:

- automatic depressurization system (ADS)
- core makeup tank (CMT)
- passive residual heat removal heat exchanger (PRHRHX)
- in-containment refueling water storage tank (IRWST)

As part of the AP600 design review, the NRC staff evaluated the ability of the LOFTRAN code to evaluate the performance of these components during transients and accidents without exceeding the capabilities of the code. Much of this review also applies to using the code for the AP1000 design review.

Westinghouse will not use the LOFTRAN code to analyze events that would cause the ADS to open, with the exception of inadvertent opening of a single valve. The scope of this analysis will be limited to the initial few seconds so that core heat transfer can be evaluated. The analysis will be terminated before significant steam voiding can occur in the reactor system.

Similarly, Westinghouse does not expect to use the LOFTRAN code to analyze conditions for which two-phase voiding will occur in the CMTs. The CMTs are expected to actuate during transients that are analyzed using the LOFTRAN code, but steam formation in the CMT inlet lines is not expected so that the CMTs will not drain in LOFTRAN analyses. Actuation of the CMTs creates a circulation path so that the cooler borated water from the CMTs mixes with the reactor coolant. This provides for additional core cooling and boron addition to shut down the core even before the CMTs begin to drain. In the event that CMT draining is calculated, Westinghouse proposes to impose a penalty model on the results. That penalty model would assume that only steam exists in the vertical piping above the CMT, thereby reducing the natural circulation driving potential. The staff did not perform a detailed evaluation of the penalty model as part of the AP600 review, since CMT draining was not calculated to occur for the AP600 standard plant design.

Of the events listed in Table 2, steam voiding in the reactor system and CMT draining would be most likely following an MSLB. This is because the accompanying reduction in steam pressure would cause a rapid increase in the rate of heat removal from the reactor system. Cooling of the reactor system would cause a pressure reduction, which might cause the CMTs to drain. A large pressure reduction in the reactor system would produce an “S” signal that would cause the RCPs to trip. Loss of forced circulation might result in elevated local temperatures causing steam formation in the reactor vessel head, CMT pressure balance lines, and intact SG tubes. The SGs for the AP1000 standard plant design are significantly larger than those of the AP600 design, and they have the potential for more pressure reduction in the reactor system than do those of the AP600. Thus, voiding might be produced within the AP1000 design following an MSLB, although none was predicted for the AP600 design. Flow restrictions in the SG nozzles are the same for the AP600 and AP1000 designs, so the rate of reactor system pressure reduction would be approximately the same. The total water mass is greater for the AP1000 SGs, so the total pressure reduction might be greater for the AP1000 design. The NRC staff requested that Westinghouse perform an MSLB analysis for the AP1000 design (see RAI P32). Westinghouse responded that it will not perform the MSLB analysis to evaluate possible reactor system voiding until the Phase 3 review for the AP1000 standard plant design. The NRC staff will therefore, defer its review and approval of the LOFTRAN code for an MSLB analysis until the Phase 3 review.

The PRHRHX provides a passive means of decay heat removal that can be effective at all reactor system pressures. As in the AP600 standard plant design, the PRHRHX for the AP1000 is located within the IRWST and transfers heat from the RCS to the IRWST for conditions when the normal means to reactor heat removal might be lost. For both designs, the PRHRHX tube bundle is C-shaped and makes a single pass within the IRWST. By contrast, the LOFTRAN code describes the IRWST as a one lumped parameter region, and the analysis does not include local heating of the IRWST water in the vicinity of the PRHRHX. The NRC staff originally had concerns during the AP600 review that higher local temperatures might cause the actual PRHR heat flow to be lower than that predicted by the LOFTRAN code. Westinghouse demonstrated, however, that the PRHRHX models in the LOFTRAN code were adequate for the AP600 design by modifying the code so that the code predications compared well with the results from scale model tests.

The PRHRHX for the AP1000 standard plant design is essentially the same as in the AP600 design, with the exception that the heat transfer area has been increased by 22 percent and the flow resistance in the inlet and outlet piping has been decreased so that the design heat flow rate is increased by 72 percent. Also, the average heat flux for the AP1000 design is expected to be 41 percent higher than for the AP600. Westinghouse uses the LOFTRAN code to calculate heat transfer through the PRHRHX using standard convective heat transfer correlations for the water flowing on the inside of the PRHRHX tubes. These correlations were found to be valid for a wide range to test conditions, including those expected for the AP1000 standard plant design. On the secondary side of the PRHRHX, the most significant heat transfer mode will be nucleate boiling. Westinghouse had previously found that standard nucleate boiling correlations overpredict the heat flow from the PRHR test facility (Reference 25). Consequently, Westinghouse modified the nucleate boiling heat transfer correlation used in the LOFTRAN code to provide a “best fit” to the data. Further verification for the derived nucleate boiling correlation was obtained by correlating PRHRHX data from the ROSA and SPES facility experiments that were performed for the AP600 design. The NRC staff therefore believes that the PRHRHX model in the modified LOFTRAN code will be valid for the AP1000 standard plant design. When Westinghouse performs individual analysis for the non-LOCA transients using the LOFTRAN code, the uncertainty in PRHR heat transfer is taken into account. Uncertainties which were determined from the scatter in the test data are included to be most conservative for the transient analyzed.

3.1.2 Conclusion

Given the forgoing considerations, the NRC staff concludes that use of the LOFTRAN code as described in References 4, 5, and 6 is acceptable for licensing calculations of the AP1000 standard plant design, subject to the following condition and limitation:

- The transients and accidents that Westinghouse proposes to analyze with the LOFTRAN code are listed in Table 2 of this report, and the NRC staff’s review of LOFTRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification.

- The NRC staff requested that Westinghouse perform MSLB analyses for the AP1000 standard plant design. In particular, the staff wanted to assess the ability of the code to model the resulting steam formation in the reactor coolant loops. Westinghouse responded that an MSLB analysis to evaluate possible reactor system voiding will not be performed until the Phase 3 review of the AP1000 design. The NRC staff will therefore defer its review and approval of the LOFTRAN code for an MSLB analysis to the Phase 3 review.

3.2 NOTRUMP Code Applicability

The NOTRUMP code was first submitted for NRC review in November 1982. The code was developed to better address the thermal-hydraulic aspects of a postulated SBLOCA, which had become an issue following the accident at Three Mile Island. The NRC staff's review indicated that the NOTRUMP code is acceptable for the analysis of SBLOCA events for Westinghouse reactor designs (Reference 26). For NOTRUMP evaluations, an SBLOCA is considered to be a rupture in the RCP boundary with a total cross-sectional area less than 1.0 square foot for which the normal charging system flow is not sufficient to maintain pressurizer level and pressure. The NRC staff has also approved the use of the NOTRUMP code for an SBLOCA evaluation for plants designed by Combustion Engineering (Reference 27).

The NOTRUMP code models one-dimensional thermal-hydraulics using control volumes that are interconnected by flow paths (links). The spacial and time-dependant solution is governed by the integral forms of the conservation equations in the control volumes and flow links. The thermal-hydraulics account for non-equilibrium thermodynamics and apply drift flux models for calculating relative velocities between the steam and liquid phases. Reactivity feedback is modeled with point kinetics neutronics. The code also incorporates special models to calculate responses of the RCPs, steam separators, and core fuel pins. A significant code feature is a node stacking capability for calculating a single mixture height in a subdivided vertical region. A two-phase horizontal stratified flow model is also included.

3.2.1 Summary of AP600 Evaluations of the Use of the NOTRUMP Code

With the AP600 standard plant design, Westinghouse introduced new systems and protective features for which the NOTRUMP code had not previously been evaluated. Specifically, these include the ADS, CMTs, PRHRHX, and IRWST. Westinghouse investigated the capability of the NOTRUMP code to evaluate the AP600 systems as discussed in Reference 28. That investigation revealed that the existing code was adequate regarding most features of the AP600; however, the following modifications were required:

- implementation of the Simulator Advanced Real-time Code (SIMARC) drift flux methodology

- general drift flux model modifications:

  - modification of the Yeh drift flux correlation for use with the SIMARC drift flux method

- inclusion of general droplet flow correlation when void fractions are between 0.95 and 1.0 when using the improved TRAC-PF1 flow regime map
- modification of the bubbly and slug flow distribution parameter (C_0)

- use of a net volumetric flow-based momentum equation
- implementation of the EPRI/flooding vertical drift flux model
- modifications to allow over-riding of the default NOTRUMP contact coefficient terms for formation of regions
- implementation of internally calculated liquid reflux flow links
- implementation of a mixture level overshoot model
- modification of the bubble rise/droplet fall model logic
- activation of the simplified pump model
- implementation of the implicit fluid node gravitational head model
- implementation of the horizontal leveling model
- implementation of the revised unchoking model
- implementation of a revised condensation heat link model
- implementation of the Zuber critical heat flux model
- revision of the two-phase friction multiplier logic
- addition of the Henry-Fauske/HEM critical flow correlation
- improvement of the fluid node staking model logic
- revision of the iteration method for transition boiling correlation in metal node heat links

Reference 29 describes these modifications, and verifies the code by comparison of calculational predictions to test data. That verification involved the use of both integral-effects tests utilizing simulated reactor systems and separate-effects tests modeling individual components. The NRC reviewed and approved the application of the NOTRUMP code for analysis of SBLOCA events for the AP600 passive reactor design (Reference 7), subject to the following conditions, which also apply to the AP1000 standard plant design:

- Westinghouse did not predict core uncovery for any design-basis SBLOCA event for the AP600 standard plant design and, therefore, did not calculate that transition boiling or film boiling might occur in the core. As a result, the NRC staff did not review the changes in the numerical solution techniques used in the NOTRUMP heat links to evaluate this condition. The staff therefore concluded that this methodology may not be invoked in applying the NOTRUMP code to the AP600 calculations. Should the NOTRUMP code be applied to calculations for which this methodology is being invoked, the NRC staff will revisit the need to review the modified transition boiling correlation solution scheme.

- The staff noted that the NOTRUMP code cannot calculate the effects of non-condensable gases injected into the primary coolant system during the AP600 SBLOCA. Non-condensable gases enter the PRHR late in the transient, when the PRHRHX no longer has a significant role in heat removal. Thus, the non-condensable gases do not appear to have a significant effect on the course of the event. If scenarios are found that cause non-condensable gases to reach the PRHRHX while it is actively removing heat from the primary system, the NOTRUMP code cannot be used to analyze those scenarios. Westinghouse removes consideration of PRHRHX heat flow prior to the ADS-4 actuation, which should prevent non-condensable gas from the accumulators from reaching the PRHR while it is included in the NOTRUMP model.

- The NOTRUMP code does not model the momentum flux terms in the conservation of momentum equation dealing with the effects of area and density changes. Westinghouse evaluated the effect of the omitted momentum flux terms and concluded that they were of little significance, with the exception of flow in the ADS-4 after the reactor system pressure decreases to a point where the flow velocity is no longer sonic. To account for this deficiency and deficiencies in the ability of the code to calculate pressurizer drainage and reactor vessel downcomer level, Westinghouse imposed a reduction in the IRWST level. This reduction conservatively delays the time of IRWST injection and produces a net reduction in the available volume of IRWST water. By comparison with data from the APEX facility, adjusted to account for scale, a reduction of 3 feet in the IRWST level was determined to be appropriate. For added conservatism, Westinghouse used an IRWST level reduction penalty of 6 feet. Westinghouse has committed to provide similar evaluations for the AP1000 standard plant design (Reference 6).

- The NRC staff questioned the ability of the NOTRUMP code to adequately predict liquid entrainment in branch lines. The most significant example occurs during ADS-4 operation. Flow through the ADS-4 valves exits the reactor system hot legs from tees located at the top of the hot leg piping. The NOTRUMP code assumes that entrainment will occur when the mixture level in the hot legs reaches a preset elevation, which is independent of the ADS-4 flow velocity. For very high ADS-4 fluid velocities, the NOTRUMP code may underpredict the amount of liquid entrained in the ADS lines. The resistance to vapor flow through an ADS-4 inlet line is reduced without entrained liquid, and this may result in vapor flow rates through the ADS-4 that are too high, in turn resulting in an excessively high rate of reactor system depressurization. For the AP600 standard plant design, Westinghouse accounted for this effect with the IRWST level penalty derived through comparisons with the APEX facility test data. Westinghouse has committed to perform evaluations of liquid entrainment through the ADS-4 as part of Phase 3 evaluation of the AP1000 standard plant design (Reference 6).

3.2.2 Evaluations of the NOTRUMP Code for the AP1000 Design Review

References 4 and 6 describe the use of the NOTRUMP code for the AP1000 evaluations. Reference 4 contains a general description of the AP1000 standard plant design and preliminary analyses of SBLOCAs in three postulated locations. Reference 6 provides additional details and justifications for using the NOTRUMP code for AP1000 SBLOCA analysis.

In Reference 5, Westinghouse provided a PIRT for an SBLOCA at an AP1000 plant, as well as an AP1000 scaling assessment of the tests that were used to qualify the NOTRUMP code for the AP600 design review. Westinghouse concludes that the NOTRUMP code is qualified to perform SBLOCA analyses for the AP1000 standard plant design without further modifications.

The AP1000 design is essentially a larger version of the AP600 design, which includes changes to increase the core power, core power density, and capacity of the passive safety systems. The design modifications that are of primary significance for modeling SBLOCA events include increasing the size of the CMTs, IRWST, ADS-4 valves, and PRHRHX. Specifically, the core length was extended to 14 feet, the thermal power output was increased by approximately 76 percent and the average linear power was increased from 4.10 to 5.707 kw/ft. The design of the RCPs has also been modified, and the SGs are larger. In addition, the RCPs are tripped on a safety injection signal, and neither the primary coolant pumps nor the SGs are expected to have a significant influence on the course of an SBLOCA event at an AP1000 plant.

The accumulators used in the AP1000 design are the same size as those used in the APP600 design, and the CMTs are 25 percent larger than for the AP600. Depressurization by the first three ADS stages will be slower for the AP1000 design, since the plant is larger but the valves for the first three ADS stages are the same size as in the AP600 design.

The design power increase of 76 percent places greater reliance on the ADS-4 to depressurize the plant so that injection from the IRWST can begin and refill the core. The ADS-4 total vent area is increased by 76 percent. The resistance to flow in the inlet lines from the hot legs to the ADS-4 valves is decreased so that the total ADS-4 relief capacity is increased by 89 percent. Instead of the IRWST level penalty that was used in AP600 analyses to account for deficiencies in the NOTRUMP ADS-4 model, Westinghouse proposes to use an increased resistance model in the NOTRUMP code for the AP1000 evaluations. In applying this model to the AP600 design, ADS-4 subsonic flow resistance was increased by 60 percent. The increased resistance was derived from correlation of the APEX facility data for a 2-inch SBLOCA and scaled for the AP600 design. Westinghouse plans to use the same resistance penalty for the AP1000 analyses, and the company believes that this approach conservatively bounds the lack of a momentum flux model in the NOTRUMP code.

Westinghouse did not perform separate stand alone tests for the ADS-4, but concludes that the NOTRUMP code was qualified to predict the ADS-4 flow rates by successful correlation of data from the APEX and SPES integral-effects test facilities, which included models of the ADS-4 valves that were scaled for the AP600 design. The NRC staff is concerned that higher steam velocities that would occur in the AP1000 upper plenum and into the horizontally stratified hot legs would invalidate the Westinghouse scaling conclusions (See Section 2.3.2 of this report).

Westinghouse proposes to benchmark its NOTRUMP predictions against a revised version of WCOBRA-TRAC, which would in turn be benchmarked against appropriate SBLOCA test data. The ability of the NOTRUMP code to adequately predict liquid entrainment in the upper plenum and hot legs and removal of liquid and vapor by ADS-4 is a concern to the staff since the amount of entrainment will affect the ability of the ADS-4 to depressurize the reactor and the amount of liquid inventory in the reactor vessel. Westinghouse proposes to address all of these issues as part of the WCOBRA-TRAC benchmark, which will be part of the Phase 3 review. See the Westinghouse response to RAI P38. Prediction of flow through the ADS-4 remains an open issue for the NOTRUMP code.

In the AP1000 standard plant design, the PRHRHX is essentially the same design as for the AP600. The heat transfer area has been increased by 22 percent and the flow resistance in the inlet and outlet piping has been decreased to increase the design heat flow rate by 72 percent. The average heat flux for the AP1000 reactor is expected to be 41 percent higher than for the AP600. In the AP600 review, the NRC staff accepted the PRHRHX model because the heat transfer that the NOTRUMP code calculated for the SPES facility and APEX facility experiments was lower than that measured in the experiments. PRHR heat transfer is given a medium importance in the PIRT (Reference 5), and is of greater importance for very small breaks since most of the reactor decay heat would be removed by a larger break.

In Reference 29, Westinghouse concludes that the NOTRUMP PRHR model contains a deficiency that needs to be monitored to ensure that excess PRHR heat transfer is not calculated. This is because the NOTRUMP code does not model the thermal plume in the IRWST that would occur as a result of extended operation of the PRHRHX. For the AP1000, design, the heat flux from the PRHRHX will be greater than for the AP600 and, therefore, more likely to produce a thermal plume in the IRWST. Westinghouse further states in WCAP-14807 that for PRHRHX flow velocities greater than 1.5 ft/sec, the heat transfer from the PRHR will be overpredicted because the secondary-side heat transfer coefficient is overpredicted because of the effect of the thermal plume on the boiling process. The flow velocity within the PRHRHX tubes of the AP1000 may exceed 1.5 ft/sec under natural circulation conditions, since the resistance to flow in the inlet and exit lines to the PRHR has been reduced. The NRC staff's SER for the AP600 (NUREG-1512) concludes that the NOTRUMP PRHR model is acceptable because the PRHRHX heat transfer calculated by NOTRUMP for the SPES and APEX facilities tests is lower than that measured in the experiments. The PRHRs for the SPES and APEX test facilities were scaled for the AP600, which has lower flow velocities and heat fluxes than does the AP1000. Westinghouse addressed these concerns in its responses to RAIs P23, P55, and P72; however, those responses are inconsistent with WCAP-14807. In the response to P72, Westinghouse states that the nucleate boiling correlation used in NOTRUMP (Thom) is dependant on the IRWST saturation water temperature and, therefore, is unaffected by the thermal plume. WCAP-14807 states that the NOTRUMP code overpredicts PRHR heat transfer because of the action of a thermal plume in the IRWST suppressing the boiling process. Figure 11-1 of WCAP-14807 clearly shows that the Thom nucleate boiling correlation is non-conservative in comparison to the test data.

WCAP-15644 (Reference 6, p. 3-20) states that should the PRHR flow velocity be higher than 1.5 ft/sec for any "significant period of time," the calculation of the limiting case (minimum mass or highest PCT) would be repeated with the PRHR heat transfer surface area reduced by 50 percent to account for potential heat transfer overprediction by the NOTRUMP code. The correction is to account for the non-conservative calculation of nucleate boiling heat transfer coefficients. RELAP5 calculations by the NRC staff predict that the PRHR flow rate will be greater than 1.5 ft/sec for the majority of the time following a postulated SBLOCA for the AP1000 design. The NRC staff requires that Westinghouse define and justify what is considered to be a "significant period of time" to trigger a reduction in PRHR surface area and to justify that a 50-percent reduction of heat transfer area is conservative given comparisons with data appropriate for the AP1000 design.

The CMTs for the AP1000 standard plant design are about 25 percent larger than for the AP600, but they are expected to perform in a similar fashion. Following a LOCA, the CMT outlet valve will open to provide makeup water to the reactor core. The opening of the CMT outlet valve will also cause relatively cool borated water to circulate from the CMTs into the reactor vessel. As the reactor system becomes voided, the CMTs will drain and provide cooling for the reactor core. Westinghouse compared the NOTRUMP predictions with data from two series of stand alone tests, that were designed to model CMT behavior in both circulation and drainage modes. That comparison revealed that the NOTRUMP code predicted that the injected fluid would be at a higher temperature than the test data, and the predicted time of injection was usually delayed. These modeling results are conservative, and the NRC staff approved the use of the NOTRUMP CMT model for the AP600 design. Since the CMTs of the AP1000 design are similar and designed to perform the same function as for the AP600, the NRC staff finds the model acceptable for the AP1000.

3.2.3 NRC Staff Audit Calculation

The NRC staff contracted to have RELAP5 input developed for the AP1000 standard plant design (Reference 30). The model was developed from an existing RELAP5 input model for the AP600 and modified to describe the AP1000 using plant data supplied by Westinghouse. As a confirmatory analysis, the staff evaluated a postulated 2-inch diameter break in a reactor system cold leg and compared the results to the NOTRUMP analysis in Reference 4. In both analyses, the break was initiated from 102 percent of full power, and the failure of one of the four ADS-4 valves was assumed. Decay heat was set at 20 percent greater than the standard promulgated by the American Nuclear Society (ANS) (Reference 31). The containment pressure was set to atmospheric. The NRC staff uses RELAP5 as an aid in understanding and evaluating the sequences and phenomena in postulated reactor accidents. RELAP5 is not a design-basis licensing tool. Conclusions regarding the acceptability or unacceptability of SBLOCA events for the AP1000 design will be founded on Westinghouse's calculations using the NOTRUMP code or another Westinghouse methodology, and not on results from RELAP5. For the postulated 2-inch cold leg break, both RELAP5 and NOTRUMP predicted that the reactor core would remain covered with water or a two-phase mixture of steam and water. Both codes also predicted that the reactor core would become highly voided, but did not calculate any core heatup. RELAP5 predicted that the core void fraction would exceed 90 percent for several hundred seconds. The NOTRUMP calculation predicted that the core void fraction would be slightly lower than that predicted by RELAP5. RELAP5 predicted depressurization of the reactor system at a faster rate than NOTRUMP in the original calculation given to the staff (Reference 4). Westinghouse stated that this discrepancy resulted from the use of old AP1000 design data in the NOTRUMP calculation. Westinghouse therefore reran the NOTRUMP calculation for the 2-inch cold leg break using more current AP1000 design information. These results were more similar to those obtained using RELAP5. Table 3 compares the sequence of events calculated by the two codes. An NRC staff examination of the revised Westinghouse results indicates that Westinghouse may be using different reactor and safeguards system trip setpoints than those used in the RELAP5 calculations. The staff has requested all current reactor and safeguards trip setpoints for the AP1000 standard plant design so that a valid comparison of the results from the two codes may be obtained.

The NRC staff analyzed a postulated double-ended DVI line rupture. In this calculation, RELAP5 predicted a small amount of core uncovery following ADS-4 actuation. The amount of core uncovery calculated by RELAP5 was minimal at a time when considerable cooling of the reactor core had already occurred. Even though some fuel cladding heatup was predicted, the cladding did not reach its original operating temperature. Westinghouse performed an analysis of a double-ended DVI line break as reported in Reference 6, and did not calculate any core uncovery. The staff did not make a detailed comparison of the results from the two codes for the DVI line break since Westinghouse stated that the NOTRUMP analyses in Reference 4 utilize old AP1000 design data.

Assessments by an NRC contractor, Information Systems Laboratory, Inc. (ISL), using standalone drift flux models indicated that a small degree of core uncovery might exist for the 2-inch cold leg break conditions calculated by RELAP5 (Reference 30). ISL believes that even if some core uncovery did occur, the amount of fuel cladding heatup would be minimal. This is because the decay heat power density is relatively low at the top of the core.

The accuracy of a computer code in predicting core uncovery can be demonstrated by the ability of the code to correlate experimental test data. Westinghouse successfully used the NOTRUMP code to correlate experimental test data for core uncovery at low pressures and for conditions that would be present in an AP1000 plant following an SBLOCA. The test data used for this correlation were from the Westinghouse G2 series of tests and the ACHILLES tests in England. Chapter 4 of Reference 29 and the Westinghouse response to RAI P69 both discuss the findings from these comparisons. In general, the NOTRUMP predictions were more conservative (predicted more core uncovery) than the test data. Consequently, the NRC staff considers the NOTRUMP code to be acceptable for calculating the two-phase level for the AP1000 design at this phase of the AP1000 review. However, the NRC staff considers the amount of low pressure core uncovery and two-phase level swell test data available for computer code verification to be very limited. The NRC staff is considering obtaining and correlating additional test data as part of the verification of the TRAC-M code (an NRC-sponsored thermal-hydraulic computer code that is currently under development). Based on the results of this developmental work, the NRC staff may revisit the calculation of core uncovery by NOTRUMP during Phase 3 of the AP1000 review.

At the present time, only a limited number of SBLOCA sizes and locations have been investigated. During Phase 3 of the AP1000 review, the NRC staff will require Westinghouse to investigate a complete small-break spectrum. If core uncovery is calculated as Westinghouse performs a complete break spectrum, the NOTRUMP calculation will be supplemented with a fuel pin heatup calculation using the Westinghouse SBLOCTA code (The SBLOCTA code is used to calculate peak cladding temperatures to show compliance with 10 CFR 50.46). Westinghouse has not submitted the SBLOCTA code for staff review for either the AP600 or the AP1000. If Westinghouse calculates core uncovery during Phase 3 of the AP1000 review, the NRC staff will require that NOTRUMP and SBLOCTA be qualified for the predicted conditions. The information required to qualify the codes has been transmitted to Westinghouse.

3.2.4 Conclusions

The NRC staff is continuing its review of the NOTRUMP code for analysis of SBLOCA events for the AP1000 standard plant design. The discussions in the preceding sections state that the NRC staff has detemined that many code features are acceptable for use in the AP1000 analysis. Nonetheless, final code approval will be contingent on the resolution of the following open issues:

- The ability of the NOTRUMP code to adequately predict liquid entrainment from the upper plenum and from stratified water in the hot legs into the ADS-4 is a concern to the staff since the amount of entrainment will affect the ability of the ADS-4 to depressurize the reactor and will affect the reactor vessel liquid inventory. Westinghouse proposes to address these issues as part of a WCOBRA-TRAC benchmark, which would be conducted as part of the Phase 3 review. Prediction of flow through the ADS-4 remains an open issue for the NOTRUMP code.

- Westinghouse has stated that the NOTRUMP PRHRHX model contains a deficiency that produces non-conservative results for high heat flows. High heat flows are identified by a velocity through the primary side of the PRHR tubes of greater than 1.5 ft/sec for any “significant period of time.” Westinghouse proposes to reduce the PRHR heat transfer area in NOTRUMP by 50 percent during these periods. The flow velocity for the AP1000 design may exceed 1.5 ft/sec for much of the time during an SBLOCA event. The NRC staff therefore requires that Westinghouse define and justify what is considered to be a “significant period of time” to trigger a reduction in PRHR surface area and to justify that a 50-percent reduction of heat transfer area is conservative given comparisons with data appropriate for the AP1000 design.

- Westinghouse has not provided a complete small-break spectrum for the AP1000 standard plant design, but proposes to submit a complete break spectrum during Phase 3 of the review. Core uncovery may be predicted for certain small break sizes and locations. Westinghouse has not provided the NRC staff with justification for using either the NOTRUMP code or the SBLOCTA code, which is used to evaluate peak cladding temperatures for the AP1000 conditions when the core is uncovered. If Westinghouse calculates core uncovery during Phase 3 of the AP1000 review, the NRC staff will require that both NOTRUMP and SBLOCTA be qualified for the predicted conditions.

3.3 Applicability of the WCOBRA/TRAC Code

WCOBRA/TRAC is a “best-estimate” analysis code used for LBLOCA and LTC analyses. The staff’s evaluation of this code for the AP600 LBLOCA and LTC analyses was documented in Sections 2.1.6.3 and 2.1.6.4 of NUREG-1512. The staff’s evaluation for its application to the AP1000 design is discussed in Sections 3.3.1 and 3.3.2 below.

3.3.1 Large Break LOCA

The NRC conducted a full-scope review of the WCOBRA/TRAC Code, which was previously approved as the LBLOCA code for the AP600 standard plant design and as a “best-estimate” code for an LOCA analysis for conventional Westinghouse plants (Reference 32). During the AP600 review, the NRC staff gained considerable understanding of WCOBRA/TRAC modeling and approximations and, thus, the staff agrees with Westinghouse’s statement in WCAP-15613 (Reference 5) that scaling from the AP600 to the AP1000 is not required. The reason is that the passive cooling system does not participate in the LBLOCA portion of the transient that is covered by WCOBRA/TRAC. Following the LBLOCA blowdown, the core is reflooded and the fuel temperature increase is terminated by the accumulator injection. The AP1000 LBLOCA recovery is similar to that of the AP600 and, thus, there are no new phenomena and no need for additional data or AP1000 scaling. However, the limitations described in Section 21.6.3.17 of NUREG-1512 for application of WCOBRA/TRAC for the AP600 LBLOCA analysis should also apply for the AP1000.

3.3.2 Long Term Cooling

The LTC performance of a reactor depends critically on the performance of the ADS-4, which has a dual role. Specifically, the ADS-4 completes the depressurization phase for the initiation of the IRWST injection, and it ejects a steam-water mixture during the LTC phase of the transient to prevent boron precipitation and accumulation in the vessel. The performance of the ADS-4 valves during the LTC phase is the subject of this review.

The AP1000 flow path pipe diameter is 14 inches (18 inches nominal) off the hot leg in a vertical direction. The hot leg diameter is 31 inches. The corresponding values for the AP600 are 10 inches (12 inches nominal) for the flow path pipe and 31 inches for the hot leg. Water entrainment in the ADS pipe depends on steam velocity, hot leg water level and flow regime, while possible de-entrainment in the pipe depends on the pipe diameter and flow regime.

Regarding the LTC phase, Section 2.3.3 of WCAP-15644 (Reference 6) states that “The simulation in WCAP-14776 predicting the APEX [facility] tests validate and justify the application of WCOBRA/TRAC to the AP1000 design certification long term cooling ECCS performance analysis.” However, it is not clear to the staff how WCAP-14776, “WCOBRA/TRAC OSU Long-Term Cooling Final Validation Report” (Reference 33), “validates and justifies” the application of the code to the AP1000 standard plant design. The AP1000 pipe diameter that feeds the ADS-4 has been increased from 10 inches in the AP600 design to 14 inches in the AP1000 design, and water entrainment in the 14-inch pipe has not been discussed in relation to the steam velocity and water level in the hot leg.

In response to RAI P019, Westinghouse states, “Inasmuch as the scaling basis of the OSU facility remain valid for the AP1000 design, this validation basis, which was approved for AP600, remains adequate for AP1000.” The “AP1000 PIRT and Scaling Assessment” in WCAP-15613 for the sump injection phase of the LTC treats only the discharge function of the ADS-4, and does not address liquid entrainment through the ADS-4 venting. Extension of the AP600 validation in WCAP-14776 to the AP1000 needs to be treated in the scaling analysis (or elsewhere) in view of the increased diameter and the potential for different flow regimes.

In response to RAI P74, Westinghouse states that “The AP1000 boron concentrations are the same as the AP600... The means by which the AP600 prevents excessive boron buildup has been extensively discussed with the NRC staff (refer to AP600 RAI 440.663).” However, the response to 440.663 is not relevant to the AP1000 without analysis or additional data.

The staff finds that Westinghouse needs to address water entrainment in the ADS-4 during sump injection in the context of the increased ADS-4 diameter and the AP1000 decay heat, steam velocity in the hot leg, steam velocity in the ADS-4 feed pipe, and flow regime. It seems physically plausible that at low decay heat, the larger ADS-4 diameter may not discharge water (because of de-entrainment at low steam velocities), and the claim that the AP1000 safety-related systems are designed to provide adequate cooling of the reactor indefinitely (WCAP-15612, Section 3.3.3.1) needs to be quantified and/or restated.

3.4 WGOTHIC Code Applicability

Westinghouse-GOTHIC (WGOTHIC) is a thermal-hydraulic computer program used for the design-basis licensing analysis of the AP600 and AP1000 passive containment designs. The WGOTHIC computer program is used to conservatively calculate the containment thermal-hydraulic response to mass, momentum, and energy releases from postulated pipe break scenarios (e.g., design-basis LOCAs and MSLBs). Westinghouse uses WGOTHIC in a lumped parameter fashion to evaluate the pressure and temperature response of the passive containment to design-basis accidents (DBAs). WGOTHIC is documented in a series of Westinghouse topical reports (References 34, 35, and 36).

3.4.1 WGOTHIC Code Description

WGOTHIC is a modified version of the GOTHIC containment analysis computer program (References 37, 38 and 39). The WGOTHIC additions include a special multi-compartment heat structure component, referred to as the “clime” model, used to model the passive containment cooling system (PCS). The essential features of the PCS include the containment steel shell, the large PCS water storage tank, the weirs on the upper containment dome for flow distribution, and the air flow path consisting of a downcomer, riser, and chimney region.

The PCS acts to reduce pressure during a DBA by removing energy through the containment shell. It also reduces pressure through the compliance (i.e., the change in energy storage resulting from a change in pressure) of the gas within the large containment volume and through heat transfer to in-containment structures. These two mechanisms are essentially the same as in existing large, dry PWR containments. However, existing containment designs also have active engineered safety features (sprays, fan coolers, and sump coolers) to remove heat to the ultimate heat sink. The passive containment design does not include active safety-grade heat removal systems. The PCS is unique and, therefore, its performance is central to this evaluation.

The primary mechanisms for heat transfer through the containment shell are condensation on the inside of the shell, conduction through the shell, and evaporative cooling on the outside of the shell. Water is released at a controlled rate and flows down the outside of the containment shell, where it is heated and evaporated. The vapor formed during the evaporation process is carried away in the air flow through the downcomer, riser, and chimney flowpath.

The staff’s evaluation concerning the application of WGOTHIC to the AP600 passive containment design is provided in NUREG-1512 (Reference 7). On the basis of that evaluation, the staff determined that the WGOTHIC computer program, combined with the conservatively biased AP600 evaluation model, was acceptable for the evaluation of the AP600 peak containment pressure following a DBA.

Although the WGOTHIC code itself is essentially a “best-estimate” tool, the evaluation methodology (EM) used by Westinghouse in support of the AP600 design certification is a conservative approach. The AP600 WGOTHIC EM uses conservative values which bound the range of most inputs, and applies conservative multipliers on the correlations used for PCS heat and mass transfer. In particular, the AP600 WGOTHIC EM uses conservative models to address the following areas:

- lumped-parameter network representation
- non-condensable circulation and stratification
- PCS flow and heat transfer models

During the peak pressure period (up to 1,200 seconds for a LOCA, and up to 600 seconds for an MSLB), these conservatisms compensate for the uncertainties introduced by the use of passive safety features, leading to an overall conservative result for the calculated peak containment pressure.

In combination with the AP600 EM, the staff has approved WGOTHIC for evaluating the peak containment pressure resulting from DBA events in the AP600. However, the program has not been qualified to predict other parameters of design interest, such as flooding levels, temperature profiles, and non-condensable concentrations (e.g., air, hydrogen).

On the basis of its evaluation, the staff identified the following limitations and restrictions for use of the WGOTHIC computer program:

On the basis of its evaluation, the NRC staff found that the WGOTHIC computer gives users great flexibility when selecting inputs. As a result, reviewers of future WGOTHIC AP600 EM analyses to support licensing actions must verify the following conservatisms in the EM model:

- The mass and heat transfer coefficients on the inner containment vessel surface are multiplied by a factor of 0.73. Only free convection is considered on the inner surface. The multiplier is based on an assessment of the large-scale test (LST) and separate effects.

- The mass and heat transfer coefficients on the outer containment vessel surface are multiplied by a factor of 0.84. Mixed convection is considered on the outer surface. The multiplier is based on an assessment of the LST and separate effects.

- The vessel wall emissivity values are reduced by 10 percent to reduce the radiation heat transfer.

- The maximum passive containment cooling water storage tank (PCCWST) temperature allowed by the technical specifications (TSs) is used as an initial condition.

- The maximum containment air temperature and maximum internal pressure allowed by the TSs are used as initial conditions. A zero percent humidity initial condition is used to increase the initial stored energy inside containment.

- A single failure of one out of two valves controlling the PCS cooling water flow is assumed. This assumption provided the minimum PCS liquid film flow rate.

- The water coverage is based on the “Evaporation Limited” flow model and the wetted surface areas.

- The minimum PCCWST inventory, as allowed in the TSs, is used to calculate the PCS flow rate for use in the “Evaporation Limited” flow model.

- The PCS liquid film flow is credited following a delay period necessary to establish water coverage of the shell-wetted region. This corresponds to the time needed to establish a steady liquid film coverage pattern in the liquid film based on the initial flow rate.

- A 20-mil or larger air gap is assumed between the steel liner and the concrete on applicable internal heat sinks. The value assumed for the air gap must be justified.

- The loss coefficient in the external annulus should include a 30-percent increase over the value derived from the test program.

- Condensation and convection on heat sinks in the dead-ended compartments, below the operating deck, should not be credited after the blowdown period. This conservative assumption should also be employed for MSLB analyses.

- Heat transfer to horizontal, upward-facing surfaces which may become covered with a condensation film is not credited. In particular, the operating deck itself, which becomes covered with an air-rich layer, should not be credited.

- For each calculation with significant energy transfer to the PCS through the shell, the stability of the “clime” heat and mass transfer solution must be examined by the combined operating license (COL) applicant (for example by plotting heat transfer rates versus time for both the wet and dry “climes”) to confirm that the calculation has not violated the time step.

- In the “Evaporation Limited” flow model, Westinghouse neglects PCS runoff sensible heat, which is conservative and offsets the non-conservatism introduced by the simultaneous use of the Chun and Seban and “Evaporated Limited” flow models. Therefore, these two assumptions must be employed together for the staff to consider this model to be acceptable for licensing analyses.

- The two-dimensional enhancement to the “Evaporation Limited” flow model may not be used to credit leakage reduction for siting evaluations.

3.4.2 Application for the AP1000

Westinghouse has requested a pre-certification review for the AP1000 design (References 4, 5, and 6) to evaluate the applicability of the use of the AP600 computer programs and test databases used to support these programs for the AP600 standard plant design.

In its original submittal (Reference 4), Westinghouse provided scoping calculations that were not consistent with the approved models and methodology developed by Westinghouse for use of the WGOTHIC computer program and approved by the staff for licensing evaluations. Westinghouse also provided the results of unverified studies using the approved modeling approach in response to the staff's RAIs (Reference 40).

The NRC staff conducted a detailed review of the material provided by Westinghouse and made a number of observations concerning that information. The following items summarize the issues that the staff raised during the review and the manner in which Westinghouse addressed the issues:

- The containment model initially presented by Westinghouse (Reference 4) for the AP1000 standard plant design differed significantly from the approved WGOTHIC model for the AP600 in the approach used to calculate LOCA mass and energy release rates, the noding strategy, and the PCS flow conditions. The double-ended cold-leg guillotine (DECLG) LOCA mass and energy release data used in the preliminary AP1000 model used an approach that was less conservative than that used in the AP600 analysis, and the methodology used to evaluate heat transfer through the PCS had been changed (Westinghouse did not use the "evaporated flow model;" instead, the company applied the full PCS flow at the top of the containment dome). These changes precluded use of the experience and insights gained by the NRC staff during the review of the AP600 model. Consequently, the staff requested (RAI P009) that Westinghouse present results for an AP1000 WGOTHIC model similar in approach to the approved AP600 model as the initial basis for discussion to avoid restarting the review process.

- The results of the scoping evaluation (Reference 4), showed that the calculated peak containment pressure of 70.7 psia was 3.0 psia less than the design pressure (73.7 psia) for the MSLB. The results also showed that the calculated peak containment pressure of 60.7 psia was 13.0 psia less than the design pressure (73.7 psia) for the DECLG LOCA, given 5 hours to release the energy from the SG secondary side. On the basis of a sensitivity study with 1 hour to release the energy from the SG secondary side (consistent with the approved methodology), the peak DECLG LOCA pressure (Reference 4) was approximately 71 psia, with a margin to the design pressure of approximately 2.7 psi.

- In Reference 40, Westinghouse provided a revised, but unverified, containment analysis that followed the nodalization approach in the region above the operating deck used for the AP600 design review. These results show the peak calculated pressure for both the MSLB and the DECLG LOCA to remain below the design pressure.

- Westinghouse has not yet provided LOCA results using the conservative mass and energy release approach that was used for the AP600 design review, nor has the company submitted comparison calculations using the "evaporated flow" model. In addition, the current analyses are based on the ADS-4, IRWST, and sump characteristics developed for the AP600 design.

- Westinghouse also clarified (Reference 41) an issue that the staff raised (RAI P003) regarding the dome area used for heat transfer. Specifically, Westinghouse noted that the same dome area of 5,200 ft^2 was used for both the AP600 and the AP1000 analyses. The information that 115 ft^2 had been used for the AP600 analysis was based on a typographical error in the information provided to the NRC; the data entry in the surface area table was the external temperature value (115°F).

- The PIRT report (Reference 5) did not sufficiently describe the expert review process. In RAI PO15, the NRC staff requested that Westinghouse provide a summary of the experts' reasoning behind no changes at the "component or volume" level as used in Table 2.6-1 of WCAP-15613. Westinghouse provided (Reference 41) two letters from Professor Per Peterson and S. G. Bankoff as an indication of the considerations given to the PIRT process. The letters provide some insight into the process used by the experts in the PIRT process. The overall conclusion was that the differences between the AP600 and the AP1000 plants are modest to small and can be treated in the analysis.

- In response to the staff's concerns regarding errors in the GOTHIC manuals (RAI P012) and also improvements and error corrections in recent versions of GOTHIC (RAI P013), Westinghouse provided information (Reference 41) regarding these issues. As part of the AP600 review and certification, Westinghouse provided clarifications regarding discrepancies in the GOTHIC manuals, and the issues regarding these discrepancies were resolved as a part of the AP600 certification. The WGOTHIC version used by Westinghouse for the AP1000 design review is based on the same GOTHIC code used for the AP600 review.

- Regarding the correction of a number of errors and deficiencies in the GOTHIC code, Westinghouse stated that its procedures require evaluation of identified errors. The potential impact of any errors that could affect the results of safety analyses will be covered in a revision to the WGOTHIC documentation (Reference 6). In regard to improved physical models in newer versions of GOTHIC, Westinghouse has demonstrated the conservatism of the models that are presently used in WGOTHIC, so use of newer models is unnecessary.

- In RAI P007, the staff noted that the test data for the chimney did not cover the range of Grashof and Reynolds numbers for the AP1000 standard plant design. Westinghouse noted (Reference 41) that the company conservatively does not use the "clime" heat and mass transfer correlations in the chimney region, so the range of test data is, therefore, not relevant. Table 4 compares the expected AP1000 range to the test data range for the PCS riser/downcomer region.

- Westinghouse used the AP600 scaling study to support the AP1000 review. However, the staff and Westinghouse agreed during the AP600 review that the LST was not properly scaled for transient situations. The LST is only valid for steady-state conditions, as acknowledged by the evaluation of PIRT. In response to RAI P014 (Reference 41), Westinghouse clarified that the LST is not well-scaled for either AP600 or AP1000. However, the LST does support the mass and heat transfer correlations used in the WGOTHIC code for the AP600 and AP1000 designs.

- In support of its determination that there is no need to account for new phenomena for the AP1000 (RAI P011), Westinghouse provided a considerable amount of information regarding the need to rewet a surface that has been heated above the saturation temperature. For the AP600, the PCS film temperature was calculated to increase to over 200°F, but was not predicted to reach the boiling point. For the AP1000, there was a question as to whether the surface temperature would reach the saturation temperature before the PCS achieved full coverage for the LOCA. Given the results of the analyses presented by Westinghouse, the staff could not conclude that containment shell temperatures would not exceed 212°F prior to full water coverage. In Reference 40, Westinghouse provided analyses which demonstrated that full water coverage would be achieved for the LOCA prior to the exterior shell temperature reaching 212°F. Westinghouse also showed that PCS flow was not needed to keep the peak pressure below the design pressure for the MSLB. For the LOCA, Westinghouse first showed that full coverage would be achieved for the AP1000 prior to the 337-second time period assumed in the analysis, based on the AP600 results. A containment shell transient heatup analysis was also performed to determine the exterior shell temperature at 337 seconds, assuming a constant 270°F containment atmosphere temperature. The results showed that the outer shell surface would reach a maximum temperature of 180°F at 337 seconds, well below the saturation temperature of 212°F.

  The 337-second time period used for the AP1000 calculation was based on the AP600 design. Westinghouse has provided an analysis which shows that the delay time for the AP1000 is less than 337 seconds (Reference 40). The actual time will be further evaluated for use in the AP1000 Phase III submittal.

- The larger height of the AP1000 (compared to the AP600) could cause more complex recirculation patterns, thereby influencing mixing. Less homogeneity of the containment atmosphere above the operating deck could result, with higher temperatures in the upper dome. However, Westinghouse should conservatively predict the decreased homogeneity using a multi-node model.

The staff has performed preliminary analyses for the AP1000 with the CONTAIN 2.0 computer program. The AP1000 design pressure is 73.7 psia. The mass and energy rates were provided by Westinghouse (Reference 42). The base CONTAIN 2.0 model and analyses do not include the conservative aspects of the Westinghouse model. For the MSLB, the staff calculated a peak pressure of 69.4 psia (compared to the Westinghouse value of 70.7 psia) (Reference 40). The DECLG LOCA mass and energy rates were based on 5 hours to release the energy from the SG secondary side. The base case DECLG LOCA results with CONTAIN 2.0 was about 54 psia (compared to the Westinghouse value of 60.7 psia) (Reference 40). A sensitivity study was performed for the DECLG LOCA to mimic the mass and heat transfer model multipliers used by Westinghouse with a resulting peak pressure of about 56 psia. Overall, the magnitude and timing of the pressure responses for both the MSLB and DECLG LOCA between the two methods is comparable. Based on Reference 5, Westinghouse appears to calculate a higher, more conservative peak temperature for the LOCA case.

3.4.3 Conclusions

On the basis of its review of the materials submitted by Westinghouse, including responses to RAIs, the staff has concluded that the WGOTHIC computer code, when applied using the methodology adopted for the AP600 review, appears to be applicable to the AP1000 standard plant design. A number of issues were raised concerning potential phenomenological differences introduced by the larger scale and higher power level of the AP1000, including whether the AP1000 shell temperature would reach boiling temperature prior to full PCS water coverage. Westinghouse provided information, including calculations of peak containment pressure for the MSLB and LOCA using a WGOTHIC model similar to that used for the AP600 review, to substantiate the applicability of the code for the AP1000 standard plant design.

The final boundary conditions for the mass and energy release rates from the MSLB and LOCA accidents; for the ADS-4, IRWST, and sump mass and energy release rates; and for the evaporated PCS flow rates will need to be reviewed to ensure that the mass and heat transfer correlations used in the WGOTHIC computer program remain valid for the AP1000 licensing calculations.

4.0 SUMMARY AND CONCLUSION

As part of the AP1000 Phase 2 pre-application review, the staff has evaluated the applicability of the AP600 test program and analysis codes to the AP1000 standard plant design. On the basis of that evaluation, the staff finds that with the exception of those items discussed below, the experimental data produced by the AP600 separate-effects and integral-effects test programs are appropriate for verification of the processes expected in an AP1000 plant, and the analysis codes validated for the AP600 standard plant design are applicable to the AP1000 design.

- As discussed in Section 2.4 of this report, none of the integral-effects facilities are sufficiently well scaled so that they provide an acceptable database to validate thermal-hydraulic codes for liquid entrainment that is expected to occur in an AP1000 plant during ADS-4 and IRWST injection periods of an SBLOCA. Westinghouse has not demonstrated that the existing AP600 integral-effects tests provide data over the range of conditions necessary to validate entrainment models in the codes that they intend to use. In particular, as discussed in Section 3.2.2, the NOTRUMP code lacks justification for modeling entrainment in the upper plenum or from a horizontal stratified water level in the hot legs. These models impact the ADS-4 flow and liquid entrainment.

- As discussed in Section 3.1.1, the NRC staff has deferred its review of the ability of the LOFTRAN code to evaluate the steam voids that might form within the reactor system following an MSLB until Phase 3 because Westinghouse does not plan to analyze the MSLB for the AP1000 design until the Phase 3 review.

- As discussed in Section 3.2.2, Westinghouse needs to qualify the heat transfer area penalty factor used with the NOTRUMP PRHRHX model. Existing PRHR test data show that the boiling heat transfer correlation used in the NOTRUMP code is non-conservative at high heat fluxes. The difference between the correlation predictions and test data becomes significant for the PRHRHX heat fluxes predicted for the AP1000 design, which are larger than those predicted for the AP600.

- As discussed in Section 3.2.3, Westinghouse has not provided the NRC staff with justification for using either the NOTRUMP code or the SBLOCTA code, which is used to evaluate peak cladding temperatures for the AP1000 conditions, when the core is uncovered.

- As discussed in Section 3.3.2, Westinghouse has not provided justification that the increased ADS-4 diameter in the context of the AP1000 steaming rate and water level in the hot legs will support liquid expulsion to avoid boron precipitation in the vessel during LTC.

- As discussed in Section 3.4.2, Westinghouse needs to perform the WGOTHIC containment analyses with an evaluation model and appropriate boundary conditions to ensure that the mass and heat transfer correlations remain valid for the AP1000 design.

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Table 1

Major Differences Between the AP1000 and AP600 Designs

SYSTEMS/COMPONENTS	AP1000	AP600
Overall Plant		
Net Electric Output, MWe	1090	600
Hot Leg Temperature, ^{o}F	615	600
Core		
Core Power, MWt	3400	1933
Number of Fuel Assemblies	157	145
Active Fuel Length, ft	14	12
Average Linear Power, kW/ft	5.707	4.10
Steam Generators		
Model	Delta-125	Delta-75
Heat Transfer Area/SG, ft^2	125,000	75,180
Number of Tubes/SG	10,000	6,307
Reactor Coolant Pumps		
Rated HP/Pump, hp	6,000	3,500
Rated Head, ft	350	240
Pump Inertia, ib-ft^2	15,750	4,956
Rated Flow/Pump, gpm	75,000	51,000
Pressurizer		
Total Volume, ft^3	2,100	1,600
Volume/MWt, ft^3/MWt	0.615	0.825
Containment		
Free Volume, ft^3	2.07E+6	1.71E+6
Safety Injection		

SYSTEMS/COMPONENTS	AP1000	AP600
Core Makeup Tanks		
Volume/CMT, ft^3	2500	2000
In-Containment Refueling Water Storage Tank		
Minimum Water Volume, ft^3	78,900	74,500
Minimum Water Height, ft	28.58	27
Available Driving Pressure, psi	9.72	9.04
Injection Line Size, inches	8	6
Injection Line to Sump Tee Size, inches	10	6
Injection Line Resistance, %	32	100
Passive Residual Heat Removal System		
Heat Changer Number of Tubes	689	671
Heat Exchanger Heat Transfer Area, ft^2	5278	4326
PRHR inlet/outlet line diameter, inches	14	10
PRHR Flow Path Resistance, %	33	100
Automatic Depressurization System		
ADS-4 Squib Valve Diameter, inches	14	10
ADS-4 Hot Leg Off-Take Pipe Diameter, inches	18	12

Table 2

Non-LOCA Transients to be Analyzed Using LOFTRAN

Feedwater system malfunctions
Excessive increase in steam flow
Inadvertent opening of a steam generator relief or safety valve
Steamline break
Inadvertent operation of PRHRHX
Loss of external load/turbine trip/MSIV closure
Loss of offsite power
Loss of normal feedwater flow
Feedwater line rupture
Loss of forced reactor coolant flow
Locked reactor coolant pump rotor/sheared shaft
Control rod cluster withdrawal at power
Dropped control rod cluster/dropped control bank
Startup of an inactive reactor coolant pump
Inadvertent actuation of the CMTs during power operation
Inadvertent increase in coolant inventory
Inadvertent opening of a pressurizer safety valve, relief valve, or ADS valve
Steam generator tube rupture

Table 3

Two-Inch Cold Leg Break Comparison Chart

Event	Time (seconds)	
	RELAP5	NOTRUMP
Break Initiates	0.0	0.0
Reactor Trip Signal	9.4	55.5
“S” Signal	52.4	62
Reactor Coolant Pump Trip	67.5	67.2
CMTs Begin to Drain	1280	1000
Accumulator Injection Begins	1642	1447
ADS-1 Actuates	2097	1337.1
ADS-2 Actuates	2177	1467.1
ADS-3 Actuates	2297	1587.1
ADS-4 Actuates	2906	2490.1
Accumulators Empty	3020	1983
Voids Above Core > 90%	3156	
CMTs Empty	3580	2890
IRWST Injection Begins	3621	3300
Max. Voids Above Core ~95%	3660	

Table 4

Comparison of Containment Dimensionless Parameters for AP1000 and Test Data

Parameter	Test Data Composite	AP1000 Estimated Range (Reference 41)	WGOTHIC AP1000 Calculation (Limiting Location) (Reference 42)
Riser Region			
Reynolds number	<120,000 (evaporation) <500,000 (dry)	<210,000	150,000
Grashof number	$<7.o\times10^{10}$ (evaporation) $<1.0\times10^{11}$ (dry)	$<1.5\times10^{9}$	1.1×10^{9}
Prandtl number	0.72 - 0.90	0.72 - 0.90	0.80
Schmidt number	~0.52	~0.52	0.55
Downcomer Region			
Reynolds number	<500,000	<190,000	103,000
Grashof number	$<1.0\times10^{11}$	$<2.1\times10^{10}$	5.2×10^{9}
Prandtl number	~0.72	~0.72	0.72